ROSA-III Experimental Program for BWR
LOCA/ECCS Integral Simulation Tests

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ROSA-III Experimental Program for BWR
LOCA/ECCS Integral Simulation Tests

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Abstract

This is the final report of the ROSA-III experimental program, in which the summary of integral simulation test results is described on thermal-hydraulic behavior during a loss-of-coolant accident (LOCA) of a boiling water reactor (BWR) and on the effectiveness of the emergency core cooling system (ECCS). Also presented in the report are the assessment results of computer codes for the BWR LOCA analysis and of the similarity between ROSA-III test results and thermal-hydraulic phenomena during a BWR LOCA by using ROSA-III test data and code analysis results.

The ROSA-III facility is a volumetrically scaled (1/424) BWR system with an electrically heated core consisting of four half-length bundles. Many test series were conducted between April 1978 and March 1983. The similarity between a ROSA-III test and a BWR LOCA concerning the fundamental thermal-hydraulic phenomena has been confirmed for major ROSA-III tests. The accident scenario has been well understood and defined for various break locations and break sizes. The effectiveness of the current BWR ECCS design has been well demonstrated.

Keywords: BWR, LOCA, ECCS, Integral Simulation Tests, Thermal-Hydraulics, Reactor Safety, Code, Similarity
ROSA-IIIによるBWR冷却材喪失事故に関する
総合模擬実験研究

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要　旨

本論文は ROSA-III実験計画の最終報告書であり、沸騰水型原子炉（BWR）の冷却材喪失事故（LOCA）時の熱水力挙動、及び非常用炉心冷却系（ECCS）の効果に関する総合模擬実験の結果についてとりまとめている。また BWR LOCA 解析コードによる計算結果の実験データとの比較にもとづくコード評価の結果、及び本実験結果の実機への適用性に関し、ROSA-III実験の結果と BWR LOCA の熱水力挙動との類似性を ROSA-III実験データと計算コードによる解析結果をもとに評価した結果について述べている。

ROSA-III実験装置は BWR を 1/424 の体積比で模擬しており、電気加熱の模擬炉心は実長の半分の長さのバンドル4体で構成されている。1978年4月から1983年3月にかけて多くのシリーズの実験がなされた。そして ROSA-III実験と BWR LOCA との基本的熱水力挙動の類似性が主要な ROSA-III実験に対して確認された。また、破断位置及び破断面積を様々に変えた場合の事故の推移が充分理解でき、現行の BWR の ECCS の有効性を実証した。
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<td>ADS</td>
<td>Automatic depressurization system</td>
</tr>
<tr>
<td>ANS</td>
<td>American Nuclear Society</td>
</tr>
<tr>
<td>AV</td>
<td>Air-actuated valve</td>
</tr>
<tr>
<td>BE</td>
<td>Best estimate (model or code)</td>
</tr>
<tr>
<td>BU</td>
<td>Break unit</td>
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<tr>
<td>BWR</td>
<td>Boiling water reactor</td>
</tr>
<tr>
<td>CA</td>
<td>Chromel-Alumel (thermocouple)</td>
</tr>
<tr>
<td>CCFL</td>
<td>Counter-current flow limiting</td>
</tr>
<tr>
<td>CHF</td>
<td>Critical heat flux</td>
</tr>
<tr>
<td>CSNI</td>
<td>Committee on Safety of Nuclear Installations</td>
</tr>
<tr>
<td>CV</td>
<td>Control valve</td>
</tr>
<tr>
<td>DC</td>
<td>Downcomer</td>
</tr>
<tr>
<td>DEB</td>
<td>Double-ended break</td>
</tr>
<tr>
<td>DG</td>
<td>Diesel generator</td>
</tr>
<tr>
<td>DISB</td>
<td>(Pump) discharge line break</td>
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<tr>
<td>DL</td>
<td>Elevation from the bottom of pressure vessel</td>
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<tr>
<td>DNB</td>
<td>Departure from nucleate boiling</td>
</tr>
<tr>
<td>DO</td>
<td>Dryout</td>
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<tr>
<td>DP</td>
<td>Differential pressure</td>
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<tr>
<td>DTT</td>
<td>Drag-disk turbine transducer</td>
</tr>
<tr>
<td>ECC</td>
<td>Emergency core coolant</td>
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<tr>
<td>ECCS</td>
<td>Emergency core cooling system</td>
</tr>
<tr>
<td>EL</td>
<td>Elevation (from the bottom of core)</td>
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<tr>
<td>ESF</td>
<td>Engineered safety features</td>
</tr>
<tr>
<td>ESTA</td>
<td>Eighteen-Degree Sector Test Apparatus</td>
</tr>
<tr>
<td>FIST</td>
<td>Full Integral Simulation Test (Facility)</td>
</tr>
<tr>
<td>FP</td>
<td>Fission product</td>
</tr>
<tr>
<td>FS</td>
<td>Full scale</td>
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<tr>
<td>FW</td>
<td>Feedwater</td>
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<tr>
<td>FWF</td>
<td>Feedwater flashing</td>
</tr>
<tr>
<td>FWL</td>
<td>Feedwater line</td>
</tr>
<tr>
<td>GD</td>
<td>Gamma-ray densitometer</td>
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<tr>
<td>GE</td>
<td>General Electric Company</td>
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<tr>
<td>GT</td>
<td>(Control rod) guide tube</td>
</tr>
<tr>
<td>HEM</td>
<td>Homogeneous equilibrium model</td>
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<tr>
<td>HFM</td>
<td>Henry-Fauske (critical flow) model</td>
</tr>
<tr>
<td>HPCS</td>
<td>High pressure core spray</td>
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<tr>
<td>HTC</td>
<td>Heat transfer coefficient</td>
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<tr>
<td>ID.</td>
<td>Identification</td>
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<tr>
<td>I.D.</td>
<td>Inner diameter</td>
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<tr>
<td>ISP</td>
<td>International Standard Problem (of CSNI)</td>
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<tr>
<td>JAERI</td>
<td>Japan Atomic Energy Research Institute</td>
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<tr>
<td>JP</td>
<td>Jet pump</td>
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<tr>
<td>JPDN</td>
<td>Jet pump drive nozzle</td>
</tr>
<tr>
<td>JPSU</td>
<td>Jet pump suction uncovery</td>
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</tbody>
</table>
LB  Large break (LOCA)
LOCA  Loss-of-coolant accident
LOCE  Loss-of-coolant experiment
LP  Lower plenum
LPCI  Low pressure coolant injection
LPCS  Low pressure core spray
LPF  Lower plenum flashing
LPF  Local peaking factor (for core power)
LTP  Lower tie plate
MCHFR  Minimum critical heat flux ratio
MLHR  Maximum linear heat rating
MRL  Main recirculation line
MRLB  Main recirculation line break
MRP  Main recirculation pump
MSIV  Main steam isolation valve
MSL  Main steam line
MSLB  Main steam line break
NEA  Nuclear Energy Agency (of OECD)
NSSS  Nuclear steam supply system
O.D.  Outer diameter
OECD  Organization of Economic Cooperation and Development
OR  Orifice
P  Rod temperature measurement location (No.1 through 7)
PCS  Pressure control system
PCT  Peak cladding temperature
PF  Peaking factor
Pos.  Rod temperature measurement location (No.1 through 7)
PV  Pressure vessel
RCV  Reactor containment vessel
RLU  Recirculation line uncovering
ROSA  Rig of Safety Assessment (Program)
rpm  Revolution per minute
RPV  Reactor pressure vessel
RRTF  Refill-Reflood Test Facility
QOBU  Quick-opening blowdown valve
QSV  Quick shut-off valve
SB  Small break (LOCA)
Sch  Schedule
SD  (Pressure vessel) steam dome
SEO  Side entry orifice
SHTF  Spray Heat Transfer Test Facility
SRV  Safety relief valve
SSTF  (Thirty-Degree) Steam Sector Test Facility
SUCB  (Pump) suction line break
SV  Safety valve
TBL  Two-Bundle Loop (Facility)
T/C  Thermocouple
TLTA  Two-Loop Test Apparatus (Facility)
<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
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<tbody>
<tr>
<td>TMI</td>
<td>Three Mile Island (Power Station)</td>
</tr>
<tr>
<td>TMI-2</td>
<td>Three Mile Island Unit 2 (Reactor)</td>
</tr>
<tr>
<td>UP</td>
<td>Upper plenum</td>
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<tr>
<td>UTP</td>
<td>Upper tie plate</td>
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</table>
1. Introduction

The loss-of-coolant accident (LOCA) is a major accident postulated in the safety evaluation for the light water reactors. In a LOCA, the coolant flows out of the primary cooling system to the containment due to the rupture of the primary coolant piping and as a result the coolability of the core degrades. Once the rupture of piping occurs, the reactor power is shut down by scram, however, the heat generation still continues for a long period with the decay heat produced by the fission products (FPs) in the fuel rods. Due to the loss of coolant and the decay heat in the core, the fuel rods temperature should rise to destroy the integrity of fuel rods without any engineered safety feature and FPs would be released into the containment as a result. This is a sequential event which is anticipated in a LOCA. Reactors are equipped with the emergency core cooling systems (ECCSs) to minimize the effect of a LOCA. The ECCS injects coolant into the core to limit the failure of the fuel rods by the superheat of the cladding and to maintain the rods in a coolable geometry, therefore, the release of FPs can be minimized.

In order to provide research information related to LOCA/ECCS, the Japan Atomic Energy Research Institute (JAERI) initiated the Rig of Safety Assessment (ROSA) program in 1970 and series of tests were conducted to investigate the thermal-hydraulic behavior of primary coolant in a nuclear reactor by using the experimental test facilities. The first phase of the ROSA program was the ROSA-I experimental program in which simple vessel-blowdown tests were performed to examine a critical flow phenomenon through a break\(^1\).

In 1974, the ROSA-II experimental program\(^2\) for the study of thermal-hydraulic behavior in the primary cooling system of a pressurized water reactor (PWR) during a LOCA was initiated with a test facility simulating a PWR system. The ROSA-II facility has two loops: one is an intact loop and the the other is a broken loop. Both loops have an active steam generator and a recirculation pump. The core is simulated by half-length electric heater rods with a maximum heater power of 2.35 MW. The volumetric scale factor of the facility is 1/416. A very effective alternate ECCS was invented and tested. This ECCS injects hot accumulator water into the upper plenum, ambient temperature accumulator water into the lower plenum, and low pressure injection system water into the hot legs.

Similar research program concerning a boiling water reactor (BWR) started in April 1978, as the ROSA-III experimental program. Since it is recognized that LOCA phenomena cannot be studied in a full scale BWR, a diverse range of large scale experiments are considered necessary to include all controlling phenomena and system interactions and to sufficiently challenge the resultant prediction methods. The total BWR LOCA safety research program includes a wide range of experiments (separate effect and integral system), model development (basic mechanisms, components and system response), and model assessment. The separate effects studies have continued for a number of years, such that the controlling heat transfer and hydraulic phenomena are now adequately defined. Therefore, the primary emphasis in recent years has been on realistic system response descriptions. Also the objective of the ROSA-III experimental program is to study the integral system response of a BWR during a LOCA. The ROSA-III facility is a BWR system volumetrically scaled at 1/424 with an electrically heated core designed to study the response of the engineered safety features in commercial BWR systems during a postulated LOCA. With recognition of the differences in commercial BWR designs and inherent distortions in reduced scale systems, the ROSA-III
facility was designed so as to produce the significant thermal-hydraulic phenomena that would occur in commercial BWR systems in the same sequence and with approximately the same time frames and magnitudes. The objectives of the ROSA-III experimental program are:

(1) To understand the primary thermal-hydraulic phenomena in the primary cooling system of a BWR during a LOCA. The performance of the engineered safety features, with particular emphasis on ECCS, and the quantitative margins of safety inherent in performance of the engineered safety features are of primary interest.

(2) To identify and investigate any unexpected event(s) or threshold(s) in the response of either the plant or the engineered safety features and develop analytical techniques that adequately describe and account for such unexpected behavior.

(3) To provide an experimental database required to evaluate and improve the analytical methods currently used to predict the LOCA response of BWRs.

Similar research programs for the integral simulation of a BWR LOCA and a transient were also conducted at TLTA\(^5\) and FIST\(^6\) in the USA and TBL\(^5\) in Japan. The scaling of TLTA and FIST is based upon one full-length bundle, TBL upon two full-length bundles, and the ROSA-III facility upon four half-length bundles. Therefore, the test results from these facilities are complementary to each other in the study of the thermal-hydraulic phenomena during a BWR LOCA and a transient. Special interest for the ROSA-III facility is the observation of thermal-hydraulic interactions between the multiple bundles.

A double-ended break at the recirculation pump suction line results in the fastest mass depletion from the reactor pressure vessel and it is considered one of the severest LOCAs of a BWR which may result in severe core uncover and heatup. Therefore, a large number of LOCA tests were conducted in the ROSA-III experimental program with a double-ended break at this location with a variety of ECCS conditions and initial thermal-hydraulic conditions. Following the accident at the Three Mile Island-2 (TMI-2) reactor, much of the focus for international reactor safety research was redirected toward boiling, level tracking, and other phenomena that distinguish small break LOCA scenarios in postulated PWR accidents. The ROSA-III experimental program was also extended after the TMI-2 accident to cover a wider spectrum of BWR break locations and sizes. The break area was changed from a minimal size to a largest size with a double-ended break and the break location was changed from recirculation pump suction line to recirculation pump discharge line, main steam line and jet pump drive line. Also steady state natural circulation tests were conducted extensively which are important for core cooling during small break LOCAs. The test matrix of the ROSA-III experimental program covers a wide spectrum of BWR LOCAs, therefore, it can be considered sufficient to fulfill the objectives (1) to (3) described above except for the three-dimensional thermal-hydraulic phenomena in the upper plenum which were studied in the Eighteen Degree Sector Test Apparatus (ESTA)\(^6\) and the Steam Sector Test Facility (SSTF)\(^7\).

The ROSA-III test results cannot be directly applied to a BWR because of facility scaling and its distortion, but careful examinations are necessary on the similarity between thermal-hydraulic phenomena at the ROSA-III facility and a BWR in order to apply the ROSA-III test results to a BWR. The ROSA-III facility is designed to simulate thermal-hydraulic phenomena of a BWR during a LOCA with primary scaling criteria of 1/2 height, 1/424 volume, flow rate and core power. Major scaling distortions are found in core power limitation, core material, simplified component geometries, and so on. One-half height scaling could have the largest influence on the thermal-hydraulic phenomena among scaling criteria and their distortions. Therefore, counterpart tests were conducted at ROSA-III (1/2 height) and FIST (full height) facilities for a large break, a small break, and a steam line break with equivalent test conditions, and it was clarified by the comparison and analysis of test results at the two test
facilities that the one-half height scaling of the ROSA-III facility has only a little influence on the thermal-hydraulic phenomena during a BWR LOCA. The results are described in Chapters 18 to 20. The similarity between the ROSA-III test results and BWR thermal-hydraulic phenomena is discussed more comprehensively in Chapter 21 before drawing conclusions in Chapter 22 where only the conclusions applicable to a BWR are described. In Chapter 21 the similarity is examined by comparing the code-predicted response of ROSA-III to that of the reference BWR. The ROSA-III data were used for the accuracy assessment of the computer code as well as input models and parameters before making such comparisons. These models and parameters were adjusted, when necessary, so that reasonable agreement was obtained between the predicted and measured ROSA-III responses. Such approach has proved useful for examining the applicability of the ROSA-III test data to a BWR because such computer code has been developed for the analysis of a BWR LOCA, in other words, the code should be applicable to a BWR LOCA analysis. There are typically between 350 and 900 fuel channels of square cross section in a BWR core and each fuel channel contains 60 to 64 nuclear fuel rods. The channels operate in parallel between lower plenum at the inlet and upper plenum at the outlet. The hydraulic coupling with the upper plenum has relatively low resistance, whereas significant orificing is placed between each channel and the lower plenum, which makes a computer code easily applicable to a BWR LOCA using models from single-bundle separate effect test results except for the three-dimensional thermal-hydraulic phenomena in the upper plenum.

In this report the ROSA-III facility is described in Chapter 2. In Chapters 3 through 17 the ROSA-III test results are given for each test series as shown below.

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<th>Chapter</th>
<th>Break Location</th>
<th>Break Area</th>
<th>Contents</th>
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<td>MRP suction</td>
<td>200%</td>
<td>Double-ended break (DEB) tests with different ECCS conditions</td>
</tr>
<tr>
<td>4</td>
<td>&quot;</td>
<td>200%</td>
<td>DEB tests with different core inlet flow rates</td>
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<td>5</td>
<td>&quot;</td>
<td>0-200%</td>
<td>Break area parameter tests</td>
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<td>6</td>
<td>&quot;</td>
<td>5%</td>
<td>OECD/NEA CSNI ISP-12</td>
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<td>7</td>
<td>&quot;</td>
<td>1.5%</td>
<td>Small break tests with different ADS condition</td>
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<td>8</td>
<td>&quot;</td>
<td>0-5%</td>
<td>Small break tests with and without pressure control system</td>
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<tr>
<td>9</td>
<td>&quot;</td>
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<td>Natural circulation tests</td>
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<tr>
<td>10</td>
<td>&quot;</td>
<td>5,50,200%</td>
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<td>11</td>
<td>&quot;</td>
<td>5,200%</td>
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<td>12</td>
<td>&quot;</td>
<td>15,50,200%</td>
<td>Break configuration (nozzle or orifice) sensitivity tests</td>
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<td>13</td>
<td>&quot;</td>
<td>5,200%</td>
<td>ECC temperature sensitivity tests</td>
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<td>14</td>
<td>MRP discharge</td>
<td>50-200%</td>
<td>Recirculation pump discharge line break tests</td>
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<tr>
<td>15</td>
<td>MSL</td>
<td>14-140%</td>
<td>Main steam line break tests</td>
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<tr>
<td></td>
<td>JP drive line</td>
<td>21%</td>
<td>Jet pump drive line break test</td>
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<tr>
<td>17</td>
<td>MRP suction</td>
<td>5-200%</td>
<td>Core heat transfer analysis</td>
</tr>
<tr>
<td>18</td>
<td>MRP suction</td>
<td>200%</td>
<td>ROSA-III/FIST counterpart tests</td>
</tr>
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<td>19</td>
<td>MRP suction</td>
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<tr>
<td>20</td>
<td>MSL</td>
<td>70%</td>
<td>ROSA-III/FIST counterpart tests</td>
</tr>
<tr>
<td>21</td>
<td>—</td>
<td>—</td>
<td>Examination of applicability of ROSA-III test results</td>
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</tbody>
</table>

Each chapter consists of sections of introduction, experiment description, experimental results, analysis, conclusion and references. Analysis section may be missing in some chapters. The conclusion in each chapter is drawn primarily from experimental analysis of the results of each test series described in the chapter without examining its applicability to a BWR. However, in some chapters the similarity between ROSA-III test results and BWR phenomena is discussed in the analysis section. The results of similarity study and/or computer code assessment are described in the analysis section. The conclusions obtained in the ROSA-III experimental program and considered applicable to a BWR LOCA are described in Chapter 22. In Appendix I described are the characteristics of the ROSA-III facility. In Appendix II all the tests conducted in the ROSA-III experimental program are listed and the results of the large break base test series and the small break test series are described in Appendices III and IV, respectively. These data in Appendices III and IV are described in the appendices because the test data have some drawbacks due primarily to a considerable leakage between downcomer and shroud inside in the reactor pressure vessel during each test, however, the obtained conclusions are useful and considered to be applicable to a BWR LOCA. The papers and reports published in the ROSA-III experimental program are listed in Appendix V.

References

2. Test Facility

2.1 Design Considerations

The ROSA-III facility\(^2\) is volumetrically scaled at 1/424 to a BWR/6-251 with 848 fuel bundles\(^3\) with an electrically heated core designed to study the response of the engineered safety feature (ESF) in commercial BWR systems during postulated LOCA\(\mathrm{s}\) including a small break LOCA. The primary characteristics of the ROSA-III facility are compared with those of the BWR/6-251 in Table 2.1.

The basic scaling and design objectives were to provide a test apparatus for investigating, on a real-time basis, the expected thermal-hydraulic response of the BWR core following a postulated LOCA. The capability exists in ROSA-III to establish the initial thermodynamic conditions of the reference BWR and to appropriately scale those parameters which govern the mass and energy transfer rates. This provides a real-time basis for the transient response of the test apparatus.

All BWR hydraulic systems which might significantly influence LOCA/ECC phenomena, including counter-current flow limiting (CCFL) phenomena, were included in the ROSA-III facility. The geometric configuration of the reference BWR system was also preserved in the ROSA-III. Specifically this includes the pressure vessel, appropriate vessel internals, two external recirculation loops, a feedwater supply system, a steam removal system and the ECCSs. The pressure vessel internals simulated in ROSA-III include the lower plenum, the core region, viz., the bundle and bypass regions, the upper plenum, the steam separator, the annular downcomer, and the steam dome. The relative distribution of the volumes of these regions has been preserved as closely as practical within ROSA-III compared to the corresponding equivalent distribution in the reference BWR.

The four bundle core modeling was adopted for ROSA-III in order to study the thermal-hydraulic interaction among the bundles. Thus, the core region of the facility was sized to accommodate four half-length test bundles. Consequently, this condition was a scaling constraint on the system and formed the basis for scaling the remainder of the test apparatus. Because the active core height is 1/2 of the BWR/6 core, the steam generation rate in the ROSA-III bundle is 1/2 that of the reference BWR. However, full-scale (1/1) velocity scaling (or 1/424 flow area scaling) was also considered for those portions that may control the CCFL behavior, or dynamic pressure drop, or critical discharge flow rate. Such portions include the core inlet orifice, the lower and upper tieplates, the jet pump drive nozzle and mixing region.

The capacity of electric power supply to the core is 4.4 MW which corresponds to 49% of the scaled (1/424) steady state power of the BWR/6 necessary to conserve the power density in the core at steady state. Reactor scram is assumed at the time of break initiation; however, the transient power is kept constant at the steady state power for approximately 8 s after break initiation due to the limited capacity of the power supply.

2.2 Description of Facility

The test facility consists of four major subsystems which have been instrumented such that desirable system parameters can be measured and recorded during a loss-of-coolant
2.2.1 Pressure Vessel

The pressure vessel simulates the pressure vessel of a BWR. It has a simulated core, a lower plenum, an upper plenum, an annular downcomer, a steam separator, a simulated steam dryer plate, and a steam dome. It should be noted that the jet pumps are not placed inside the pressure vessel. The downcomer gap is extremely narrow to maintain the volumetric scaling ratio and relative elevations of downcomer top and bottom. It is difficult to install extremely small jet pumps in such a narrow downcomer space, therefore, it was decided to install the jet pumps outside the pressure vessel.

The core consists of four half-length 8×8 fuel bundles heated electrically. Each fuel bundle is composed of 62 fuel rods and two water rod simulators arranged in an 8×8 square array with 16.16 mm pitch, and a channel box. Figures 2.3 and 2.4 show axial power distribution and radial power distribution, respectively. The active length of the core is 1880 mm and the outside diameter of the fuel rod is 12.27 mm. Each fuel rod is a sheathed electric heater with Nichrome as the heater element and Inconel 600 as the sheath. The rod diameter and array geometry are the same as those of the reference BWR. The power supplied to the peak power bundle (Bundle A) is 1.4 times greater than that to an average power bundle (Bundle B, C or D). The local power distribution in the bundle simulates that of the reference BWR. Each bundle has fuel rods with local peaking factors of 1.1, 1.0 and 0.875, as shown in Fig. 2.4. The simulated fuel rods are electrically heated and have a chopped cosine axial power distribution with an axial peaking factor of 1.4, as shown in Fig. 2.3. The downcomer is annular and filler blocks are used to fill the gap between the square core-shroud and the circular vessel wall and to simulate the volume and geometry of an annular downcomer.

2.2.2 Steam Line and Feedwater Line

The steam line and the feedwater line simulate those of a BWR. However, the steam line and feedwater line are independent open loops for a ROSA-III test. Steam is discharged to the atmosphere through the steam line connected to the steam dome. The steam line has three branches as shown in Fig. 2.2. The first branch has a control valve (CV-130) to control the steady state steam dome pressure before blowdown. The second branch simulates the automatic depressurization system (ADS) using an orifice with a diameter of 15.5 mm. The first branch also has an orifice (OR3) with a diameter of 18.0 mm to simulate the flow resistance of a steam turbine-generator. The third branch is not used in a test. Immediately after the blowdown initiation, the control valve CV - 130 fully opens and the flow rate under a transient condition is determined by the critical flow through the orifice OR3.

The feedwater line is connected to the feedwater sparger located above the downcomer region. The feedwater conditions are controlled by the feedwater tank (FWT) and the heat exchangers (EX1 and EX2).

2.2.3 Coolant Recirculation System

The coolant recirculation system simulates the BWR recirculation loop. The system consists of two loops provided with a recirculation pump and two jet pumps in each loop. One of recirculation loops is the intact loop which simulates the unbroken loop of a BWR and the other is the broken loop which simulates the broken loop of a BWR. The broken loop
has two break simulators and a quick shutoff valve (QSV) to simulate a double-ended shear break or a split break. Each break simulator is composed of an orifice or a nozzle which determines the break area, and a quick opening blowdown valve (QOBV). The break type, position, and area are experimental parameters. The standard break condition is a 200% double-ended shear break at the recirculation pump inlet side with an orifice diameter of 26.2 mm.

2.2.4 Emergency Core Cooling System

The ECCS of ROSA-III simulates that of a BWR. The ECCS includes the high pressure core spray (HPCS), the low pressure core spray (LPCS), the low pressure coolant injection (LPCI) and the ADS systems. The Spray systems, HPCS and LPCS, spray the emergency core cooling water on the top of the core. The LPCI system supplies the emergency core cooling water into the core-bypass region directly. Each ECCS is provided with a tank, a pump, a valve and a control system to control the valve trip delay, valve opening speed and the flow rate.

2.3 Power Curve for Simulated Fuel Rods

The power curve for simulated fuel rods in the ROSA-III test\(^4\) is shown in Fig. 2.5 comparing with the calculated results for the BWR/6 plant. The ROSA-III facility has heater rods in the core for fuel rod simulation, therefore, it is quite important for ROSA-III tests to give appropriate transient power to heater rods, since there are large differences in thermal characteristics between nuclear fuel rod and electrical heater rod. Especially there is a large difference in the heat capacity of the stored heat.

The power curve of ROSA-III simulates the heat transfer rate in a BWR/6 core neglecting the stored heat of ROSA-III heater rods. The heat transfered to the coolant in a BWR core consists of delayed-neutron fission power, decay power of fission products and actinides, and the stored heat release from the fuel. The heat transfer in a BWR core was calculated by assuming nucleate boiling throughout a transient.

For the first 50 s after the initiation of blowdown, the heat transfer rate in the BWR core was calculated with the RELAP4J code assuming a transient of a 200% double-ended break at the recirculation pump suction. Moderator density feedback, Doppler feedback, and scram worth were considered in the calculation of fission power. The ROSA-III core consists of four half-length fuel bundles corresponding to 9 MW steady state power. However, the power supply to heater rods of ROSA-III is limited to 4.4 MW, therefore, the power is kept constant for the first 7.5 s until the normalized power in the core decreases to 0.49 (= 4.4 MW/9 MW).

Between 50 and 260 s after the initiation of blowdown, the stored heat release rate was calculated assuming fluid temperature to be equal to saturation temperature. Decay power of fission products and actinides was calculated by the proposed ANS 5.1 standard\(^5\) assuming thermal neutron fission of \(^{233}\)U and production rate of \(^{239}\)U per fission to be 0.59. Delayed-neutron fission power is negligible after 50 s.

After 260 s the stored heat release also becomes negligible and the heat transfer rate to coolant was calculated by the decay power of fission products and actinides.

2.4 Instrumentation

The instrumentation system of the ROSA-III\(^6\) was designed to obtain thermal-hydraulic data during a LOCE to contribute to assess the analytical code. Table 2.2 summarizes the
number and type of the instrumentation used in the ROSA-III facility.

Measured quantities in the ROSA-III test are pressure, differential pressure, flow rate, electric power, pump revolution, signals, fluid temperature, liquid level, fluid density and momentum flux of coolant flow.

Pressure measurements are done with semi-conductor transducers measuring the piezoelectric resistance. The detector is cooled by water for the protection from high temperature environment.

Differential pressure transducers with two direct current cables convert displacement of a diaphragm to electric charge and then to proportional voltage. The pressure lead pipes are dual circular pipes for circulating cooling water to eliminate flashing of the fluid.

The flow rate is measured by orifice, venturi, turbine or electromagnetic flow meters according to the fluid condition and the measurement location.

The electric power for simulated fuel rods is controlled by the predetermined function of time and it is measured by a fast response electric power meter.

The pump revolution speed is measured by an electromagnetic counter on the pump shaft.

On-off signals such as valve position, pump revolution direction, blowdown initiation and pump power supply are converted to voltage or current and are recorded in respective channels in order to specify the exact time of the signal.

The temperatures of fluid, structure materials and fuel rods are measured with Chromel-Alumel thermocouples (CA T/C) of 1.6 mm or 1.0 mm O.D.

The liquid levels are measured by means of needle type electrical conductivity probes developed in the ROSA-III program. The probes are attached on the walls of core barrel and channel boxes at several elevations and detect the existence of water or vapor at each level.

The fluid density in the pipe is measured by means of a gamma-ray densitometer. Each gamma-ray densitometer has two or three beams to determine the flow regime. The gamma source is $^{137}$Cs (20 Ci) and the detector is a NaI scintillator which is cooled by water.

The momentum flux is measured by a drag disk, and the combination of two signals from a drag disk, a turbine meter and/or a gamma-ray densitometer is used to determine the two-phase flow condition.

The data acquisition system consists of DATAC-2000B and FACOM M200 system computer. The DATAC 2000B records measured data on a magnetic tape up to 750 channels and the data tape is processed by the FACOM M200 system computer at JAERI by off-line control. After the evaluation of each data by comparing the initial and final values with the standard values of the pressure, the data are re-processed using the correct conversion factors determined from the consistency examination.

2.5 Test Procedure

Most of experiments in the ROSA-III simulated LOCA transients. The following is the general procedure adopted to perform the experiments.

Prior to initiation of experiment, the system was filled with demineralized water and heated using core power. The core flow was controlled by changing the recirculation pump speed. The downcomer liquid level was controlled with the feedwater control valve. The system pressure was maintained constant at 7.3 MPa (BWR normal condition) using the steam control valve.

After the initial steady-state conditions were thus established, a LOCA experiment was initiated by opening the blowdown valve (s). Scram concurrent with the break was simulated.
The steady-state core power was maintained for 9 s after the break because the ROSA-III steady-state core power with summation of power distribution was 44% of the scaled BWR/6 rated power and corresponded to the scaled BWR/6 core power at 9 s after scram. Thereafter, the core power was reduced following a predetermined curve as described in Section 2.4. The recirculation pumps were concurrently tripped with the break.

The downcomer collapsed liquid level signals were used to trip the main steam isolation valve (MSIV) closure and ECCS actuation. The initial liquid level in the downcomer was 5.0 m above the pressure vessel bottom. With this liquid level, the downcomer liquid volume, including the jet pump suction line volume, was scaled at 1/424 of the BWR/6 downcomer liquid volume below scram level (L3 level). Likewise, the L2 and L1 liquid levels were determined to be 4.76 and 4.25 m, respectively. The level signals were generated based on differential pressure measurement in the upper downcomer above the elevation of 3.9 m. The MSIV closure was initiated by the L2 level signal with a 3 s delay. The ADS was tripped on by the L1 level signal with a 120 s delay. These time delays are the same as used in BWR safety analyses. The LPCS and LPCI were tripped on by the L1 level signal with a 40 s delay. The actual injection, however, was initiated at vessel pressures of 2.2 MPa for LPCS and 1.6 MPa for LPCI, respectively, to simulate the pump shutoff head of each system.

The feedwater flow was throttled at 2 s and terminated at 4 s after the break. A main steam line control valve was used to establish steady state and to simulate the transient operations of the MSIV, pressure control system, and safety relief valve (SRV). Upon the initiation of a test, the control valve was fully opened to increase the main steam line flow rate to the scaled (1/424) BWR/6 steam flow rate, because the steady-state core power was 44% of the scaled BWR/6 core power and the initial main steam line flow rate was less than the scaled BWR/6 main steam line flow rate. Then the valve was throttled again to simulate the BWR/6 pressure control system, which maintains the vessel pressure above 6.7 MPa. The valve was tripped to close by an L2 level signal with a 3 s delay to simulate the MSIV closure. Finally, the valve was used to simulate SRVs and maintained the vessel pressure below 8.1 MPa. The standard operational trip conditions are summarized in Table 2.3.

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### Table 2.1 Primary characteristics of BWR/6 and ROSA-III

(a) Major design parameters

<table>
<thead>
<tr>
<th></th>
<th>BWR6 (251/848)</th>
<th>ROSA-III</th>
<th>Ratio ($\frac{\text{BWR6}}{\text{ROSA-III}}$)</th>
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<td>Simulated BWR</td>
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<td>Number of</td>
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<td>Recirc. Loops</td>
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<td>4</td>
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<td>Jet Pumps</td>
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<td>4</td>
<td>6</td>
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<td>Separators</td>
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<td>Core Heat Generation</td>
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<td>Total Power (kW)</td>
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<td>Active Fuel Length (m)</td>
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<td>Total Volume (m³)</td>
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<td>Operating Conditions</td>
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<td>Pressure (MPa)</td>
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<td>Core Flow (kg/s)</td>
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<td>$&lt;3.64$</td>
<td>$&gt;424$</td>
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<td>Steam Flow (kg/s)</td>
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<td>$&gt;424$</td>
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<td>Recirc. Pump Flow Rate per 1 Pump (kg/s)</td>
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<td>Feed Water Temp. (K)</td>
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Table 2.1 (continued)

(b) ECCS conditions

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<td>HPCS</td>
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<td>Injection Flow Rate (m³/s)</td>
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<tr>
<td>at 7.90 MPa</td>
<td>0.104</td>
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<td>at 1.38 MPa</td>
<td>0.442</td>
<td>0.820 X 10⁻³</td>
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<td>up to 393</td>
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<td>Injection Location</td>
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<td>upper plenum</td>
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<tr>
<td>LPCS</td>
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<td>Injection Flow Rate (m³/s)</td>
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<td>at 0.84 MPa</td>
<td>0.442</td>
<td>0.94 X 10⁻³</td>
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<tr>
<td>Water Temp. (K)</td>
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<td>up to 393</td>
<td></td>
</tr>
<tr>
<td>Injection Location</td>
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<td>upper plenum</td>
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<td>LPCI (RHR)</td>
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</tr>
<tr>
<td>Number of Lines</td>
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<tr>
<td>Injection Flow Rate (m³/s/pump)</td>
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<td>at 0.14 MPa</td>
<td>0.470</td>
<td>4.27 X 10⁻³</td>
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<td>Water Temp. (K)</td>
<td>300 ~ 344</td>
<td>up to 393</td>
<td></td>
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<td>Injection Location</td>
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Note: * Design scaling ratio
Table 2.1 (continued)

(c) Volume distribution and pressure vessel dimension

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<tr>
<th>Item</th>
<th>BWR6 (251/848)</th>
<th>ROSA-III</th>
<th>Ratio (BWR6/ROSA-III)</th>
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</thead>
<tbody>
<tr>
<td>Lower Plenum &amp; Guide Tubes  m³</td>
<td>123</td>
<td>0.177</td>
<td>695</td>
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<tr>
<td>Lower Plenum m³</td>
<td>79.0</td>
<td>0.112</td>
<td>705</td>
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<tr>
<td>Guide Tubes m³</td>
<td>43.8</td>
<td>0.0651</td>
<td>673</td>
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<tr>
<td>Core m³</td>
<td>59.8</td>
<td>0.134</td>
<td>446</td>
</tr>
<tr>
<td>Core in Channels m³</td>
<td>35.4</td>
<td>0.0814</td>
<td>435</td>
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<tr>
<td>Core Bypass m³</td>
<td>24.4</td>
<td>0.0524</td>
<td>465</td>
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<tr>
<td>Upper Plenum &amp; Steam Separators m³</td>
<td>80.5</td>
<td>0.185</td>
<td>435</td>
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<tr>
<td>Upper Plenum m³</td>
<td>52.5</td>
<td>0.124</td>
<td>423</td>
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<tr>
<td>Steam Separators m³</td>
<td>28.0</td>
<td>0.0610</td>
<td>459</td>
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<tr>
<td>Steam Dome *1) m³</td>
<td>206</td>
<td>0.439</td>
<td>468</td>
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<tr>
<td>Downcomer *2) m³</td>
<td>123</td>
<td>0.233</td>
<td>528</td>
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<tr>
<td>Above Jet Pump Suction m³</td>
<td>74.2*</td>
<td>0.164**4)</td>
<td>452</td>
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<tr>
<td>Between Jet Pump Suction and Recirculation Outlet m³</td>
<td>36.8*</td>
<td>0.0690*4)</td>
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<tr>
<td>Below Recirculation Outlet m³</td>
<td>12.2</td>
<td>0.00900</td>
<td>1360</td>
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<tr>
<td>Recirculation Loops &amp; Jet Pumps m³</td>
<td>29.6</td>
<td>0.171**4)</td>
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<td>Total Volume m³</td>
<td>621</td>
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Pressure Vessel Dimension

<table>
<thead>
<tr>
<th>Item</th>
<th>BWR6 m</th>
<th>ROSA-III m</th>
<th>Ratio (BWR6/ROSA-III)</th>
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<tr>
<td>Inner Height m</td>
<td>22.3*</td>
<td>6.01</td>
<td>3.71</td>
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<tr>
<td>Inner Diameter m</td>
<td>6.38*</td>
<td>0.492**5)</td>
<td>13.0</td>
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<tr>
<td>Water Level m</td>
<td>14.1*</td>
<td>4.62</td>
<td>3.04</td>
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<tr>
<td>Jet Pump Suction Level m</td>
<td>8.28*</td>
<td>2.82</td>
<td>2.93</td>
</tr>
<tr>
<td>Lower Core End Level m</td>
<td>5.49*</td>
<td>1.60**6)</td>
<td>3.43</td>
</tr>
<tr>
<td>Recirculation Line Level m</td>
<td>3.88*</td>
<td>0.938</td>
<td>4.13</td>
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<tr>
<td>Recirculation Loop Pipe Inner Diameter m</td>
<td>0.54</td>
<td>≤ 0.0495</td>
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Note: * BWR 5
*1) above normal water level
*2) below normal water level
*3) include jet pump suction lines
*4) not include jet pump suction lines
*5) out diameter of lower downcomer
*6) bottom of active fuel
Table 2.1 (continued)

<table>
<thead>
<tr>
<th></th>
<th>BWR6 (251/848)</th>
<th>ROSA-III</th>
<th>Ratio ($\frac{\text{BWR6}}{\text{ROSA-III}}$)</th>
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<tr>
<td>Active Core Length (m)</td>
<td>3.708</td>
<td>1.880</td>
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<tr>
<td>Number of Fuel Rods</td>
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<td>Number of Water Rods</td>
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<td>Rods Array</td>
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<td>Fuel Rods O.D (mm)</td>
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<tr>
<td>Cladding Thickness (mm)</td>
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<td>Fuel Rod Pitch (mm)</td>
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<td>Total Fuel Heat transfer Area</td>
<td>$7.515 \times 10^3$</td>
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<td>Cladding Material</td>
<td>Zircalloy</td>
<td>Inconel 600</td>
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<td>Average Linear Rod Power (kW/m)</td>
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<td>$\leq 9.54$</td>
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<td>Core Average Heat Flux (kW/m²)</td>
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<td>$\geq 2.04$</td>
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<td>Core Coolant Flow Rate* (kg/s)</td>
<td>$1.54 \times 10^6$***</td>
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<td>Core Inlet Velocity** (m/s)</td>
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<td>1.11</td>
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<tr>
<td>Total Core Flow Area (m²)</td>
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<td>Peaking Factor</td>
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<td>Local P.F.</td>
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<tr>
<td>Axial P.F.</td>
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<td>Radial P.F.</td>
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<td>Gross P.F.</td>
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<td>Total P.F.</td>
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<td>2.20</td>
<td>0.955</td>
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* Include core bypass
** Exclude core bypass flow rate assuming it as 10% of core coolant flow rate
*** Current 8 X 8 fuel rods
### Table 2.2 ROSA-III instrumentation summary

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<th>ITEM</th>
<th>SENSOR</th>
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<td>Pressure</td>
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<td>ECCS Loop</td>
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<td>Venturi Flow Meter</td>
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<td>Primary Loop</td>
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<td>Orifice Flow Meter</td>
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<tr>
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<td>Recirculation Loop</td>
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<td>Orifice Flow Meter</td>
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<td>Main Steam Line</td>
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<td>Capacitance Probe</td>
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<tr>
<td>Density</td>
<td>Gamma Densitometer</td>
<td>10</td>
<td>2 Beam GD</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>3 Beam GD</td>
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<td>Drag Disk</td>
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<td>Break Orifice</td>
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<td>Revolution Counter</td>
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<td>Electric Core Power</td>
<td>VA Meter</td>
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<td><strong>TOTAL</strong></td>
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### Table 2.3 Standard operational trip conditions

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<th>Events</th>
<th>Trip Conditions</th>
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<tr>
<td>break</td>
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<tr>
<td>scram</td>
<td>time 0 s</td>
</tr>
<tr>
<td>pump trip</td>
<td>time 0 s</td>
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<td>core power reduced</td>
<td>time 9 s</td>
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<tr>
<td>feedwater terminated</td>
<td>time 2 to 4 s</td>
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<tr>
<td>MSIV closure</td>
<td>$L2(4.76 \text{ m}) + 3 \text{ s}$</td>
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<td>ADS</td>
<td>$L1(4.25 \text{ m}) + 120 \text{ s}$</td>
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<td>SRV</td>
<td>$P \geq 8.1 \text{ MPa}$</td>
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<tr>
<td>HPCS</td>
<td>$L2 + 27 \text{ s}$</td>
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<tr>
<td>LPCS $L1 + 40 \text{ s}$</td>
<td>$P &lt; 2.2 \text{ MPa}$</td>
</tr>
<tr>
<td>LPCI $L1 + 40 \text{ s}$</td>
<td>$P &lt; 1.6 \text{ MPa}$</td>
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</tbody>
</table>

*note*  $L1, L2$: Level signal  
$P$: pressure at upper plenum
Fig. 2.1 Schematic of ROSA-III test facility.

Fig. 2.2 ROSA-III flow diagram.
Fig. 2.3 Axial power distribution of heater rod.

Fig. 2.4 Radial power distribution of core.
Fig. 2.5 Normalized power transient in BWR and ROSA-III.
3. Double-Ended Break Test Series

3.1 Introduction

The double-ended break LOCA tests with the break location at the recirculation pump suction line have special importance because they simulate the design basis accidents (DBAs) with the fastest loss of coolant from the primary cooling system.

This section summarizes the results of the double-ended break LOCA tests in which various single failure assumptions were made in ECCS performance\(^1\).

The primary objectives of the present test series were

1. to provide baseline information on thermal-hydraulic behavior during a postulated double-ended break LOCA with single failure conditions,
2. to evaluate ECCS performance in cases of such single failure conditions, and
3. to assess the capability of computer codes in predicting the thermal-hydraulic behavior in a double-ended break LOCA.

The experimental results were analyzed with the RELAP4/MOD6/U4/J3 code to assess the predictive capability of the code.

3.2 Test Conditions

Five tests have been conducted with various single failure assumptions made in ECCS performance.

- Run 901: full ECCS actuation
- Run 902: LPCI-DG (diesel generator) single failure
- Run 924: LPCS-DG single failure
- Run 926: HPCS-DG single failure
- Run 905: No ECCS actuation

Runs 901 and 905 are the reference tests for comparison purposes in which all or none of the ECCSs were assumed to function, respectively. One of the three diesel power generators was assumed to fail in other three runs, resulting in the failure of two out of three LPCI systems in Run 902, failure of LPCS and one LPCI system in Run 924 and an HPCS system in Run 926\(^2\). The test conditions are summarized in Table 3.1 for the present double-ended break tests. The break location was fixed at the recirculation pump suction line and the break type was a 200% double-ended break. Blowdown was initiated by opening the quick-opening blowdown valves located immediately downstream the break nozzles.

The initial conditions of the primary loop before the break were as follows. The steam dome pressure was 7.35 MPa and the temperature corresponding to the saturation temperature of 562 K. The steady state power was 3.96 MW yielding a maximum linear heat rate (MLHR) of 16.7 kW/m. The core inlet flow rate was 16 kg/s and the core outlet quality was estimated to be 14%. The lower plenum subcooling was 11 K.

3.3 Test Results

3.3.1 System Pressure and LOCA Scenario

The system pressure transients in the test Runs 901, 902, 924, 926 and 905 are shown
in Fig. 3.1. Little difference is observed in the system pressure transients among these tests. The timing of major events summarized in Table 3.2 indicates that the major events occurred at approximately the same time into the transient in all five tests, including the base tests, Runs 901 and 905.

The system pressure decreases after break due to the discharge of fluid through the break, but soon starts to increase after closure of the main steam isolation valve (MSIV) between 6 and 9 s after break. As the mixture level in the downcomer decreases rapidly after break and falls below the outlet nozzle of the recirculation loop at 13 s after break, the steam in the vessel discharges directly through the vessel side break and the system pressure begins once again to decrease rapidly.

At about 17 s after break, the system pressure falls below 6.4 MPa, and the lower plenum fluid reaches the saturation condition and begins to flash. Lower Plenum Flashing (LPF) continues and slows down the rate of decrease in the system pressure.

The fluid in the feedwater line saturates at the system pressure of 2.14 MPa and the depressurization rate is again slowed down because of the inflow of two-phase fluid from the feedwater line to the pressure vessel.

The HPCS is actuated at 27 s after the downcomer liquid level falls to the L2 level and LPCS and LPCI are actuated at system pressures of 2.16 MPa and 1.57 MPa, respectively, in accordance with the ECCS actuation conditions.

The mixture level in the core recovers rapidly after actuation of LPCS and LPCI and the whole core is quenched shortly after reflooding.

### 3.3.2 Break Flow

The break flow rates, which are measured by combinations of gamma-densitometers and drug disks located upstream the breaks, are shown in Fig. 3.2 for the HPCS failure test, Run 926. The break flow rate on the pump side starts to decrease rapidly after break because of the decrease in the average fluid density due to initiation of vaporization at the upstream side of the break. The pressure upstream the break on the pump side decreases rapidly after break and the fluid initiates vaporization because of the pressure drop due to flow choking at the jet pump drive nozzles in the broken loop upstream of the pressure measurement location.

The vessel side break flow decreases gradually after break as the system pressure falls. Break flows at the pump side and the vessel side recover temporarily when the system pressure rises due to the closure of an MSIV. However, the break flow rates start to decrease again rapidly when the jet pump suction and the outlet nozzle of the recirculation line in the downcomer are uncovered. The uncovering results in a decrease in the density of fluid discharged through the break in both cases. The uncovering of the jet pump suction occurs a little earlier than the uncovering of the outlet nozzle of the recirculation line. However, the flow through the break on the pump side starts to decrease a little later than the flow through the vessel side break due to the lower fluid velocity in the MRP-side break flow path.

### 3.3.3 Liquid Level

The mixture level transients in the pressure vessel measured with conductivity probes are shown in Fig. 3.3 for Run 926 (No HPCS actuation). The mixture level in the downcomer falls rapidly after break due to the loss of primary coolant through the break. The jet pump suctions are uncovered at 10 s after break and the exit nozzles of the recirculation line at 12 s.

The mixture level drops into the lower plenum at 32 s after break following the initiation of LPF (lower plenum flashing), though the core is not completely dried out yet. This indicates the occurrence of CCFL at the core inlet orifice.
As the flashing in the lower plenum subsides, the mixture level in the core starts to drop at 37 and 41 s after break in the average channel C and the peak channel A, respectively. At 71 s after break, the whole core is uncovered and water in the feedwater line flashes at a system pressure of 2.14 MPa. Flashing in the feedwater line holds up the system pressure temporarily, decreases the steam generation rate in the lower plenum, and causes the CCFL break down at the core inlet orifice. Consequently, the fall in the core mixture level is accelerated as the steam flows into the top of downcomer from the feedwater line.

The mixture level in the lower plenum starts to rise slowly after actuation of LPCS at 71 s because of the inflow of LPCS water from the upper plenum to the lower plenum through the core and core bypass region. The mixture levels in the core and lower plenum start to recover rapidly at 100 s following the LPCI actuation at 96 s after break. The mixture level in the core recovers to the top of the core at 112 and 122 s in the peak power bundle A and the average power bundle C, respectively. The mixture level in the peak power bundle always stays above that in the average power bundle primarily because of the higher steam generation rate and larger level swell. The mixture level in the lower plenum decreases temporarily between 104 and 121 s, possibly because of the steam binding effect in the core. The mixture level in the lower plenum recovers to the core inlet orifice level at 137 s after break.

There is a considerable delay in the recovery of the mixture level in the downcomer, because the ECCS water is injected inside the core shroud. The presence of liquid is signaled by conductivity probes after 156 s just below the jet pump suction level in the downcomer indicating the inflow from the lower plenum through the jet pump. The jet pump suction is covered by the two-phase mixture at approximately 210 s after break.

The mixture level transients in the peak power bundle A obtained in five runs are compared in Fig. 3.4. The fall in the mixture level in Run 905 (without ECCS actuation) agrees well with the mixture level behavior in Run 926 (with actuations of LPCS and 3 LPCIs), because the whole core was uncovered to the steam environment in Run 926 before the initiation of LPCS. The mixture level recovered to the top of the core after actuations of LPCS and LPCI in Run 926 while no recovery was observed in Run 905.

The behavior of mixture level was similar in Runs 901, 902 and 924. The fall in the mixture level was accelerated temporarily in these tests due to feedwater flashing, however, the bottom part of the core remained below the mixture level in all cases. On the other hand, the whole core was uncovered in Run 926 indicating the effectiveness of HPCS for maintaining the mixture level in the core. As expected, the mixture level recovery was the earliest in Run 901, in which ECCS was fully operated. The mixture level recovery in Run 902 with actuations of HPCS, LPCS and one LPCI was earlier than that in Run 924 in which HPCS and two LPCIs were activated. Since the total ECC water flow rates were similar between those two tests and the LPCS was started earlier than LPCI, the test results show the importance of early actuation of ECCS for the mixture level recovery.

### 3.3.4 Surface Temperature of Simulated Fuel Rod

The surface temperatures of the simulated fuel rod A-11 (see Fig. 2.4(b)) in Run 926 (without HPCS) and Run 905 (No ECCS) are compared in Fig. 3.5. The rod A-11 is the peak power rod with the local peaking factor of 1.1 located at the corner of the peak power channel A.

The trends prior to the ECCS actuation are similar between the two tests. The fuel surface temperatures measured start to increase rapidly between 5 and 10 s after break due to the DNB above the midplane of the core. These fuel surface temperatures exhibit rewetting between 17 and 35 s after break because of the mixture level recovery to the top of the core
due to LPF, which started at 17 s after break and led to the improvement in core cooling.

When the LPF becomes less effective, the mixture level in the core starts to fall as previously shown in Fig. 3.3 and the fuel surface temperatures start to rise sequentially from the top to the bottom part of the core. There is a strong correlation between the fuel surface temperature transients and the mixture level transients in the core shown by a dashed line. The initiation of the rise in fuel surface temperature of the rod A-11 at a given location corresponds exactly to the time of uncovering of the fuel surface due to the fall in the mixture level in the core.

After initiation of LPCS at 71 s after break in Run 926, the fuel surface rewets temporarily in both the upper and the lower regions of the core due to improved core cooling by the water falling from the upper plenum. The whole core is finally quenched from the bottom of the core upward by reflooding after LPCI was activated at 96 s after break. On the other hand, the fuel surface temperatures keep rising in Run 905 (without ECCS actuation) except at the top part of the core (Position 1). The differences in the fuel surface temperature transients between Runs 926 and 905 noted above clearly show the effectiveness of ECCS for core cooling even with a failure of HPCS. The whole core is quenched before 180 s after break by injection of ECC water inside the core shroud from LPCS and LPCI in Run 926, whereas the fuel surface temperatures kept rising everywhere in the core in Run 905 which was run without ECCS.

The peak cladding temperatures (PCTs) obtained in the five tests are compared in Fig. 3.6 and summarized in Table 3.3. The PCT is expectedly the lowest in Run 901 with full ECCS actuation. Although the total ECC injection rate is approximately the same in Runs 902, 924 and 926, the PCT in Run 902 (with a failure of 2 LPCIs) is lower than the PCT in Run 924 (with a failure of LPCS and 1 LPCI) by 54 K. The PCT in Run 924 is in turn lower than that in Run 926 (with a failure of HPCS) by 56 K. These results show the importance of early actuation of ECCS in limiting the PCT to a lower value.

The HPCS, LPCS and LPCI were actuated at approximately 30, 70 and 95 s after break in the four tests with ECCS actuation. However, the differences in the PCTs among these tests are rather small. The lowest PCT is 654 K observed in Run 901 (with full ECCS actuation) and the highest PCT is 784 K observed in Run 926 (without HPCS actuation). Therefore, the difference between the lowest and the highest PCTs is 130 K. In contrast, the cladding surface temperature in Run 905 (without ECCS actuation) kept rising steadily, showing clearly the effectiveness and the importance of the ECCS for core cooling.

3.4 Analysis

3.4.1 Analytical Model

A post-test analysis of Run 926 (without HPCS) was performed with the code RELAP4/ MOD6/U4/33). The highest PCT appeared in Run 926 in the double-ended break test series with a single failure assumption in ECCS.

The ROSA-III system was represented by 22 volumes, 41 junctions, and 19 heat slabs in the analysis. The core was divided into two regions, one for the average power bundles and the other for the peak power bundle. Each region had 7 heat slabs simulating the heater rods. The bubble rise model was used for the volumes in the pressure vessel to calculate the mixture level in the core, downcomer, lower plenum and so on, with the Wilson’s correlation3) for bubble rise velocity and the constant velocity for the steam dome and the lower downcomer. The HEM (homogeneous equilibrium model) critical flow model was used in the discharge flow calculations with a discharge coefficient of 1.0. The CCFL model developed by Wallis3)
was applied after the initiation of LPCS for the analysis of the phenomena at the outlets of the peak power bundle, average power bundle and core bypass. The ECC flow was not reduced artificially to keep the upper plenum empty of water but a large reverse flow friction factor \(10^6\) was applied at the core exit to prevent the water from falling back from the upper plenum. The fallback flow can be calculated with the CCFL correlation.

After the actuation of LPCS, the spray heat transfer model was used. The spray heat transfer coefficient was set at 14 W/m\(^2\)K (2.5 Btu/ft\(^2\)h\(^\circ\)F) before quench and 5700 W/m\(^2\)K (1000 Btu/ft\(^2\)h\(^\circ\)F) after top-down quench by spray and bottom-up quench by reflooding\(^3\)).

When the mixture level covers the fuel surface, the heat transfer coefficient after quench is applied to the calculation of thermal response of the fuel surface. Therefore, the present model neglects the film boiling heat transfer process before quench but this simple model is adequate for the PCT calculation. A revised model for calculating the whole temperature transient will be discussed in Chapter 17. The PELAP4/MOD6 best-estimate heat transfer logic was used for the analysis of phenomena before the actuation of LPCS.

Measured data of the steam flow rate, feedwater flow rate and LPCS and LPCI flow rates were used as the given conditions in the analysis. The initial temperature and quality of each volume and the flow rate at each junction were determined to be consistent with the experimental data taken at steady state.

### 3.4.2 Analytical Results

As shown in Fig. 3.7, the calculated system pressure agrees well with the measured pressure during a rapid decrease after break, recovery after MSIV (Main Steam Isolation Valve) closure, and a rapid pressure drop after the recovery of the recirculation line outlet in the downcomer. The calculated depressurization rate slows down somewhat after the initiation of the lower plenum flashing. However, the deceleration is not so significant as in the measured result, possibly because of insufficient consideration of heat release from pressure vessel internal structures and uncertainty in break flow quality in the present calculation. The temporary pressure hold-up due to flashing in the feedwater line is not calculated because the feedwater line is not included in the analysis. However, it is considered that the overall agreement in the behavior of the calculated and the measured system pressure transients is satisfactory.

The trend of the calculated mixture level is in good agreement with the experimental results (Fig. 3.8), though there are some differences in detail. The fall in the mixture level due to mass depletion from the system by break flow and rapid recovery due to level swell by LPF are observed in the calculated results, however, the magnitude of the fall before the lower plenum flashing is much smaller than in the experiment. It is probably due to two reasons. First, uniform void distribution is assumed in the analysis which makes the void fraction smaller than the experimental value at the surface of the mixture level and makes the level fall slower. Second, Wilson's correlation for bubble rise velocity used in the analysis is not applicable in the high pressure and high void fraction range.

The whole core is uncovered to the steam environment both in the calculation and the experiment, however, the calculated mixture level starts to recover shortly after the core uncovering because the accumulation of LPCS water in the core is overestimated. The whole core is reflooded within 30 s after initiation of LPCI both in the analysis and the experiment.

The cladding surface temperature transients calculated for the peak power channel are compared with the measured results for the rod A-77 in Fig. 3.9. In the calculation, the heat-up before the lower plenum flashing is underestimated in the top region and overestimated below the center of the core. The heat transfer rate calculated before the uncovering of the cladding surface below the mixture level is smaller than the experimental data, therefore, the
calculated cladding surface temperature is higher than the measured result. The rate of increase in temperature after uncovery of the cladding surface above the mixture level is in good agreement with the measured result because the heat transfer coefficient after dryout is calculated accurately by the code.

The temperature transients after turnaround are quite different between the calculation and measurement, because a simplified heat transfer model was used in the present calculation. At the top of the core, the cladding surface temperature is calculated to decrease gradually after turnaround, because the heat transfer coefficient is averaged for the entire heat slab in the calculation though the top-down quench by LPCS occurs partially at the top heat slab in the experiment.

Between the top and the midplane of the core at Positions 2 through 4, the cladding surface temperature is calculated to rapidly decrease immediately after the reflooding process begins, because the high nucleate-boiling heat transfer coefficient is applied neglecting the low film-boiling heat transfer coefficient in the present calculation. The peak temperature is calculated to be higher than the measured temperature by 110 K, primarily because the heat transfer coefficient is underestimated below the mixture level before uncovery. Neglect of the film boiling heat transfer process during reflooding does not significantly affect the peak temperature, because the measured cladding surface temperature turns around immediately after reflooding. When the film boiling heat transfer coefficient of 140 W/m²K (25 Btu/ft²h°F) is used after bottom flooding in the analysis, the cladding surface temperature turns around to decrease as in the test but the cladding surface did not show any quench in this calculation because the present heat transfer model after LPCS actuation allows the use of only one heat transfer coefficient for bottom-up flooding.

At the bottom of the core, the cladding surface temperature is also calculated to rapidly decrease immediately after recovery with the coolant, because the heat-up above the saturation temperature is well below the minimum film boiling temperature. The calculated temperature transient is in good agreement with the measured result.

The present calculation model underestimates the heat transfer coefficient during a time interval between the lower plenum flashing and the core uncovery and overestimates the cladding surface temperature. However, the peak temperature can be calculated within ±110 K of the experimental value.

In summary, the present calculation model is considered to be capable of predicting the important features of the transient including the system pressure, mixture level in the core and the peak cladding temperature in the ROSA-III large break test, Run 926.

3.5 Conclusions

The following conclusions have been obtained from the results of the double-ended break tests conducted at the ROSA-III facility and the RELAP4/Mod6/U4/J3 analyses of the test data of Run 926 (without HPCS actuation, which resulted in the highest PCT).

1. The existing ECCS design is highly effective in cooling the core during a double-ended break LOCA and there is a large margin between the measured PCT and the present licensing criterion of 1473 K even with assumptions of a certain single failure in ECCS.
2. The HPCS failure is the severest of all the single failure assumptions for emergency core cooling even in a large break LOCA. The PCT was the highest in the HPCS failure case, followed by the single LPCS-DG failure and LPCI-DG failure. This shows the importance of the early actuation of ECCS for core cooling during a LOCA.
(3) The analysis showed that the RELAP4/MOD6/U4/J3 code is capable of predicting the important features of a ROSA-III double-ended break LOCA experiment, however, the areas in need of further improvement were also identified.

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Table 3.1 Summary of test conditions

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<td>HPCS (s)</td>
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* DG : Diesel generator

Table 3.2 Comparison of major events

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<td>132.0</td>
<td>125.0</td>
<td>131.0</td>
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<tr>
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<td>132.0</td>
<td>148.0</td>
<td>184.0</td>
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Table 3.3 Comparison of peak cladding temperatures, times and locations

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<th>RUN NO.</th>
<th>901</th>
<th>902</th>
<th>924</th>
<th>926</th>
<th>905</th>
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<td>PCT (K)</td>
<td>652</td>
<td>675</td>
<td>729</td>
<td>785</td>
<td>–</td>
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<tr>
<td>Time (s)</td>
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<td>96</td>
<td>117</td>
<td>119</td>
<td>–</td>
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<tr>
<td>Location (Rod, Elevation)</td>
<td>C13, Pos.3</td>
<td>A88, Pos.3</td>
<td>A11, Pos.2</td>
<td>A71, Pos.4</td>
<td>–</td>
</tr>
<tr>
<td>Failed ECCS</td>
<td>–</td>
<td>2LPCI</td>
<td>LPCS, 1LPCI</td>
<td>HPCS</td>
<td>LPCS, 3LPCI</td>
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</tbody>
</table>
Fig. 3.1 Comparison of system pressure transients.

Fig. 3.2 Break flow in Run 926 without HPCS actuation.
Fig. 3.3 Estimated liquid level in upper plenum, core, channel inlet, lower plenum, guide tube and downcomer in Run 926 without HPCS actuation.

Fig. 3.4 Comparison of liquid levels inside core shroud.
Fig. 3.5 Comparison of cladding surface temperature transient of a peak power rod A-11 in Run 926 (without HPCS actuation) and Run 905 (without ECCS actuation).

Fig. 3.6 Comparison of peak cladding temperatures.
Fig. 3.7 Comparison of calculated and measured system pressure transients.

RLU : Recirculation line uncovering
LPF : Lower plenum flashing
FWF : Feedwater flashing

Fig. 3.8 Comparison of calculated and measured mixture levels in high power channel.

LPF : Lower plenum flashing
LPF : Lower plenum flashing

Fig. 3.9 Comparison of calculated and measured cladding temperature transients of an average power rod A-77 in the peak power channel.
4. Effects of Initial Fluid Conditions on Large Recirculation Line Break LOCA

4.1 Introduction

Most of the BWR LOCA integral tests have been performed by establishing the same initial test conditions as the BWR rated condition. Here is a tacit understanding that a probability of an occurrence of BWR LOCA at the rated fluid condition is relatively higher than that at deviated fluid conditions due to a longer operation time period for the former.

However, there are many operating phases in a BWR life time, such as start-up, shutdown and other anticipated transients including deviation of the core flow and feedwater flow from the normal conditions. Therefore, it is valuable for better understanding of a BWR LOCA to investigate effects of the deviated initial fluid conditions from the normal conditions on the BWR LOCA phenomena.

Objectives of this chapter are to clarify the effects of following test parameters on a 200% main recirculation line break (MRLB) LOCA phenomena in the ROSA-III test facility. The test parameters are deviations of (1) initial core quality distribution and initial fluid mass inventory, which are a result of larger or smaller core flow rate, (2) initial core inlet subcooling, which is a result of larger or smaller feedwater flow rate, and (3) feedwater temperature. The meaning of the third one is to investigate the effects of flashing of feedwater retaining in the feedwater line piping, which is opened to the downcomer in a BWR system.

Two parametric tests of Runs 942 and 943\textsuperscript{11} were compared with a standard 200% MRLB LOCA test of Run 926 (see Chapter 3) with an HPCS failure assumption in order to satisfy these test objectives. The initial core outlet quality was changed from 5% to 43% including the normal BWR condition of 15%, the initial core inlet subcooling was changed from normal one of 10 K up to 21K, and the feedwater temperature was changed from the normal one of 489 K to 318 K, where no feedwater flashing was expected during the LOCA period. By comparing the test results of Runs 942 and 943 with those of Run 926, the effects of three parameters are shown.

4.2 Test Conditions

The initial and transient test conditions of Runs 942 and 943 are compared in Table 4.1 with those of Run 926, a standard 100% break test with an assumption of HPCS failure. The initial test conditions of these two tests (Runs 942 and 943) were varied significantly from the standard test (Run 926) as shown in Table 4.1 as the test parameters. On the other hand, the transient test conditions were the same among the three tests.

The measured initial test conditions of Run 942 were: core flow rate of 5.9 kg/s, core inlet subcooling of 21 K, main steam flow of 2.0 kg/s and feedwater flow of 2.1 kg/s. The initial core power corresponded to 44% of the 1/424 scaled BWR rated power. The initial core inlet flow corresponds to 16% of the 1/424 scaled BWR core flow. Therefore, the initial average fluid quality at the core outlet of Run 942 was estimated as 43% and it was fairly higher than that of BWR (15%), in spite of the larger core inlet subcooling.

The measured initial test conditions of Run 943 were: core flow rate of 34.1 kg/s, core inlet subcooling of 12 K, main steam flow of 1.4 kg/s and feedwater flow of 1.5 kg/s. The
reason of smaller steam and feedwater flows in Run 943 than the other two tests were due to
the lower feedwater temperature of Run 943. The initial core power was the same as Run 942. The
initial core inlet flow of Run 943 corresponded to 94% of the 1/424 scaled BWR core
flow. Therefore, the initial average fluid quality at the core outlet of Run 943 was estimated
as 5% and it was fairly lower than that of BWR (15%).

By changing the initial core flow and core quality distribution, the initial fluid mass
inventories of Runs 942 and 943 were also deviated from that of the standard test of Run 926.
The initial fluid mass in each region was calculated for the three tests as shown in Table 4.2
by using RELAP5/MOD1CY18\textsuperscript{23} code. The fluid mass in the table includes mass of steam.
In the two-phase and steam regions ((1) through (4) in Table 4.2), the fluid mass in Run 942
was 30% less than the standard test, whereas that in Run 943 was 50% larger than the standard
case. However, the initial fluid mass in these two-phase and steam regions in the standard
test was only 13% of the total system mass including the feedwater mass remaining in the
feedwater piping. Most of the fluid mass (87%) was in the water regions of the downcomer,
lower plenum, guide tube, core bypass, jet pumps and recirculation loops. Therefore, deviations
of the total fluid mass in Runs 942 and 943 from that of Run 926 were not so large as the
deviations of the two-phase and steam region mass mentioned above. As the initial fluid
temperature in these water region in Run 942 was lower than Runs 943 and 926, the fluid
mass in these water regions was slightly larger than those of the two tests.

In the calculation of downcomer fluid mass in each test, the actual water level was estimated by correcting the negative pressure loss\textsuperscript{1} in the initial condition, which was the results of downward downcomer flow. Namely, the pressure losses of Runs 942, 926 and 943 were 0.0, 0.07 and 0.23 m in water head, respectively (see actual initial water levels in Table 4.1). By accounting these downcomer water level corrections, the fluid mass in the water region ((5) through (9) in Table 4.2) was obtained. That in Run 942 was 5% smaller and that in Run 943 was 12% larger than the standard test of Run 926.

On the other hand, hot water remaining in the feedwater line piping which was connected
to the downcomer region, was taken into account for Runs 942 and 926 because the hot
water began to flash at the system pressure of 2.2 MPa. In Run 943, however, the feedwater
mass was not accounted in the mass inventory because the feedwater temperature was low
(318 K) enough to avoid the flashing during the transient.

Consequently, the total fluid mass in the ROSA-III system was 707 kg, 742 kg and 794
kg in Runs 942, 926 and 943, respectively. Thus, the deviation of the total initial fluid mass
in Runs 942 and 943 were −5% and +7% from the standard test of Run 926.

Figures 4.1 and 4.2 show the initial average steam quality and initial linear heat rate of
heater rods with the local peaking factor of 1.1 in the average and high-power bundles, re-
spectively. In this calculation, a 10% core bypass flow was assumed and the saturated fluid
enthalpies of steam and water in the upper plenum were used. The core outlet qualities of
average and high-power bundles in Runs 943, 926 and 942 were calculated as 4% and 8%,
13% and 20%, and 38% and 59%, respectively. The core outlet quality of Run 942 was the
maximum achievable value without causing boiling transition during the initial steady state.

4.3 Test Results

The effects of three parameters on large MRLB LOCA phenomena are shown here by
comparing the test results of (1) pressure response and timings of major events, (2) downcomer
water level and system water mass, and (3) core cooling phenomena.
4.3.1 Effects of Initial Fluid Conditions on System Pressure and Timings of Major Events

Tables 4.3(a) and (b) compare the timings of major events and peak cladding temperatures (PCTs) of the three tests, respectively. The timings of main steam isolation valve (MSIV) closure, recirculation line uncovering (RLU), lower plenum flashing (LPF) initiation, feedwater flashing (FWF) initiation and actuations of ECCSs are also shown on pressure response of each test as shown in Fig. 4.3.

Deviation of the initial downcomer water mass (−6.4% and 13.7% from the standard test) caused a slight difference of downcomer water level falling speeds between Run 942 or Run 943 and the standard test of Run 926, and also caused deviations of timings of L2 level trip, MSIV closure and RLU between the two tests. However, these time differences from Run 926 were within 4 s.

The larger initial core inlet subcooling of 21 K in Run 942 resulted in delay of LPF initiation (at 5.2 MPa), which was 6 s later from the LPF initiation at 6.4 MPa, which occurred in the standard test of Run 926. The delay of LPF initiation in Run 942 resulted in difference of pressure response as shown in Fig. 4.3. However, the depressurization rate in Run 942 after the LPF initiation became lower than the other two tests because of the slightly small steam discharge flow due to the lower system pressure. And the pressure of Run 942 became close to those of Runs 926 and 943 at the time of FWF initiation.

The lower feedwater temperature in Run 943 eliminated flashing of the remaining feedwater, which was seen in Runs 942 and 926 (Fig. 4.4). In order to clarify the FWF effects, the remaining water mass in pressure vessel (PV) was evaluated and compared with the remaining hot feedwater. In Run 942, there were mixture levels only in the downcomer bottom, guide tube and lower plenum at the time of FWF initiation and the total fluid volume under the mixture levels was obtained as 0.156 m³ (see Table 4.4). In a case of void fraction of 50%, the remaining water mass in PV becomes 66 kg and therefore the feedwater mass of 30 kg contributes to increase the total saturated water mass by 45%. This means that the remaining feedwater mass has an influential affect on the large break LOCA phenomena and therefore, it should be taken into account in BWR LOCA studies.

Moreover, stored heat in the structural metals including the thick PV wall and the downcomer filler blocks can contribute to increase the steam volume when the feedwater flows into the downcomer region and contacts the metal surfaces. In fact, the metal surface temperature at the time of FWF initiation was nearly equal to the initial temperature of 552 K, because there was little heat release from the metal wall surrounding the downcomer after the downcomer level fell to the bottom. The hot metal wall, of which temperature was 60 K higher than the saturation temperature at 2.2 MPa, can increase saturated steam volume and generate super-heated steam in the downcomer region.

Thus, the feedwater flashing delayed the depressurization by generating steam in two ways: first by adding directly the saturated steam mass to the whole system steam mass and second by increasing the release of the metal stored heat. The FWF delayed the actuation of LPC1 by 11 s. The system pressure of Run 943 was fairly lower than the other two tests in the reflooding phase because of no FWF initiation.

4.3.2 Effects of Initial Fluid Conditions on Downcomer Water Level and Fluid Mass inside Shroud

Figure 4.5 shows the downcomer water levels which were measured by the differential pressure transducers in the lower and upper downcomers and Fig. 4.6 shows the differential pressure between the top and bottom of PV in the three tests. These results are useful to
compare the fluid mass inventories in the pressure vessel in the three tests.

There were little differences in the timings of downcomer level fall of the three tests (within 4 s) due to deviation of the initial downcomer water mass, and also in the water level recovery times (within 11 s) of the three tests due to initiation of the FWF. The similar trends were observed in the differential pressure between the top and bottom of PV. The effects of initial core inlet subcooling was found in the differential pressure in Fig. 4.6 as the slight difference in the LPF initiation times in the three tests. Conclusively, the effects of initial system mass and initial core inlet subcooling on the transient system mass inventory were not prominent in the whole test periods.

4.3.3 Effects of Initial Fluid Conditions on Core Cooling Phenomena

Figures 4.7 and 4.8 show the typical heater rod surface temperatures at seven elevations in the core in Runs 942 and 943, respectively. The dryout and quench timings of heater rods (A-22, B-22, C-22 and D-22 see Fig. 2.4(b)) located at the similar position (see Chapter 2) of the four bundles and the two phase mixture level in the average-power bundle (bundle C) in Runs 942, 943 and 926 are shown in Figs. 4.9(a), (b) and (c), respectively. Figure 4.10 compares the PCT and its timing of each test. The PCT rod location of each test is shown in Table 4.3(b). By comparing these results, the common features and the effects of the deviated initial fluid conditions on the core cooling phenomena were found as follows.

The following common features of the core cooling phenomena in a 200% MRLB LOCA test (see Chapter 3) were found in Runs 942 and 943. The dryout and quench fronts of the heater rods, and consequently the heater rod temperature responses were strongly related with the two-phase mixture levels in the core. The dryout and quench fronts and the core mixture level were located higher than the collapsed water level inside the core-shroud as shown in Figs. 4.9(a) through (c) except for a short time period after the break. Two types of heater rod temperature excursions were observed commonly during the test periods. Namely, the first one was due to rapid void increase immediately after the break and was diminished or suppressed by the LPF initiation. The second one was due to mixture level fall to the bottom of the core and was diminished by the reflooding initiated after the LPCI actuation. Figure 4.10 compares the PCT and its timing of each test. The PCT location of each test is shown in Table 4.3(b).

The following are the effects of the deviated initial fluid conditions on the core cooling phenomena. The higher initial core quality and delay of the LPF initiation in Run 942 clearly affected the responses of the two-phase mixture level in the core and heater rod temperatures especially in the early blowdown phase. Namely, both the smaller initial core water mass and the delayed LPF initiation in Run 942 resulted in the earlier and more significant mixture level fall (or voiding) in the upper half core, and consequently resulted in the PCT at position 3 of A-88 rod as 901 K in the early blowdown phase (see Fig. 4.10). The PCT of Run 942 was 190 K higher than the temperature at position 3 of the same heater rod in Run 926 in the same blowdown phase. If the LPF were initiated normally at 6.4 MPa in Run 942, the turn-around times of the unquenched heater rod would be earlier by 6 s than the test results. This earlier turn-around time of 6 s corresponds to a decrease of the PCT by approximately 50 K in Run 942. Thus, the higher initial core quality in Run 942 raised the heater rod surface temperatures by 140 K compared with the standard test, and the delayed LPF initiation by 6 s in Run 942 raised the temperature by 50 K from the standard test. On the other hand, the lower initial core quality of Run 943 caused slightly higher core mixture level and lower heater rod temperatures at the upper core region compared with those in Run 926 in the early blowdown phase.
Next, temporary rewetting of the heater rods (B-22, C-22 and D-22) in the three average-power bundles was clearly observed after the LPCS actuation in Run 942 (see Fig. 4.9(a)). The main reason of the temporary rewetting can be attributed to the effect of the FWF. Namely, the FWF decreased the depressurization rate as shown previously and therefore, it reduced the steam generation inside the core-shroud (mainly in the lower plenum) in comparison with the time period before the FWF initiation in both tests. The lowered steam generation rate in the lower plenum caused CCFL break at the side-entry orifice (SEO) and also at the upper core plate (UCP) of the average-power bundle with a short time delay from the former. Therefore, some amount of the two-phase fluid remaining above the upper core plate and at the bottom of the core flowed down into the core and the lower plenum, respectively, after the FWF initiation in Run 942.

However, the rewetting of the heater rods in the high-power bundle was not so clear as those in the average-power bundles in Run 942. Moreover, only B-22 rod was completely rewetted and C-22 and D-22 rods were locally (at top and bottom of the core) rewetted in Run 926 in the same time period as in Run 942. The main reasons of these differences may be attributed to (1) difference of steam generation rates between the different power bundles, and (2) multi-bundle effects among the four bundles. And in Runs 942 and 926, dryout of heater rods were observed for the third time around the LPCI actuation time in the bundles, in which the LPCS water fell down previously. On the other hand, these temporary rewetting and the third dryout were not observed in Run 943 except for the top and bottom of the core. Thus, the different core cooling phenomena were observed among the four bundles in the reflooding phase relating with the FWF.

The effect of FWF on the PCT of Runs 926 and 943 is shown as follows. As the FWF initiation at 2.2 MPa in Run 926 delayed the LPCI actuation by 11 s, it delayed similarly the turn-around time of the heater rod temperature, which recorded the PCT in the reflooding phase. The effect of FWF on the PCT value was estimated approximately as 40 K in Run 926 by assuming the similar heater rod temperature responses after the LPCI actuation. And the estimated PCT of Run 926 became very close to the PCT of Run 943. As there was little effects of the CCFL break on the heater rod temperatures in the high-power bundle, in which the PCT occurred in Run 926, it can be concluded that the lower initial core quality of 5% results in small deviation of the PCT from that of the standard test.

All the heater rods were finally quenched from bottom of the core by the recovered water mass after the LPCI actuation in the three tests. As shown in Table 4.3, the final core quench time was 104 and 79 s after each LPCI actuation time of Runs 942 and 943, respectively, and the deviations of the final core quench time after the LPCI actuation were within 13 s from the standard test.

4.4 Conclusions

The effects of the deviated initial fluid conditions from the rated BWR conditions on a 200% MRLB LOCA were investigated experimentally. The common features of the 200% MRLB LOCA phenomena (see Chapter 3) were observed in the two tests of Runs 942 and 943 with the slight influences of the following test conditions.

The initial core outlet quality was changed from the rated BWR condition of 15% (Run 926) to 5% (Run 943) and 43% (Run 942). These deviations resulted in changes of core water mass by −30% and +50%, which contributed to deviation of the total water mass in the system by −5% and +7%. On the other hand, the core inlet subcooling was changed from the standard condition of 10 K to 12 K and 21 K. The feedwater temperature was changed
from the standard condition of 489 K to a very low value of 318 K to suppress the feedwater flashing in Run 943.

Following conclusions are obtained.

(1) The higher initial core outlet quality of 43% caused earlier heater rod temperature excursion immediately after the break and consequently a PCT of 901 K, which was approximately 140 K higher than the heater rod temperature of the standard test in the early blowdown phase. The lower initial core quality of 5% resulted in little difference in the heater rod temperature responses during the whole test period.

(2) The larger initial core inlet subcooling of 21 K delayed the initiation of the LPF by 6 s from the standard condition and consequently resulted in delay of the temporary core rewetting by the LPF. The increase in PCT due to the LPF delay was estimated as 50 K. The effect of the LPF initiation was significant on the heater rod temperatures in the early blowdown phase of the larger MRLB LOCA.

(3) The hot feedwater remaining in the feedwater pipe line began to evaporate violently at 2.2 MPa. This FWF initiation slightly decelerated the depressurization and reduced the steam generation rate in the lower plenum. This caused temporary CCFL break down at the SEO (side entry orifice) and UTP (upper tie plate), and the resulting temporary rewetting and then dryout in the average-power bundles. The FWF delayed the LPCI actuation by 11 s, which delayed reflooding and thus raised the PCT in the high-power bundle by approximately 40 K.

(4) The heater rods were cooled completely by the LPCI actuation within 104 s in the three 200% MRLB LOCA tests with an assumption of HPCS failure.

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### Table 4.1 Test Conditions of Runs 942, 926 and 943

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<tr>
<th>Items</th>
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<th>Run 942</th>
<th>Run 926</th>
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\(^{\dagger}\) Average core outlet quality excluding the core bypass flow.

\(^{\ddagger}\) Actual water level was obtained by eliminating the downcomer flow effects on the differential pressure measurement in each test. The apparent water level in Runs 942, 926 and 943 were 5.04, 5.05 and 5.04 m, respectively.

\(^{\ast}\) L1 level = 4.25 m above PV bottom.

L2 level = 4.76 m above PV bottom.
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</tr>
<tr>
<td>(8) Downcomer Level Correction(§1)</td>
<td></td>
<td>9.9</td>
</tr>
<tr>
<td>(9) Jet Pumps + Recirculation Loops</td>
<td>0.1604</td>
<td>124.0</td>
</tr>
<tr>
<td>(10) Total</td>
<td>0.8185</td>
<td>613.3</td>
</tr>
<tr>
<td>Difference from Standard</td>
<td></td>
<td>(-8.5)</td>
</tr>
<tr>
<td>(11) (4) + (10)</td>
<td>1.3919</td>
<td>666.4</td>
</tr>
<tr>
<td>Difference from Standard</td>
<td></td>
<td>(-35.6)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(-5.2%)</td>
</tr>
<tr>
<td>(12) Feedwater Line</td>
<td>0.0357</td>
<td>30.3</td>
</tr>
<tr>
<td>(13) Total System (11) + (12)</td>
<td>1.4276</td>
<td>706.6</td>
</tr>
<tr>
<td>Difference from Standard</td>
<td></td>
<td>(-35.6)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(-4.8%)</td>
</tr>
</tbody>
</table>

(§1) The actual water level in the downcomer was corrected by subtracting a pressure loss term caused by the downcomer flow as 5.04 m, 5.12m and 5.27 m in Runs 942, 926 and 943, respectively. The excess mass on 5.00 m level was calculated.

(§2) There was no feedwater flashing in Run 943.
### Table 4.3 Major events and Peak cladding temperatures (PCTs) of three tests

(a) Major events

<table>
<thead>
<tr>
<th>Events</th>
<th>Time (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Run 942</td>
</tr>
<tr>
<td>Break Initiation</td>
<td>0.0</td>
</tr>
<tr>
<td>L2 Trip in Downcomer Level</td>
<td>1.6</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>4.2 to 8.8</td>
</tr>
<tr>
<td>L1 Trip in Downcomer Level</td>
<td>5.2</td>
</tr>
<tr>
<td>Recirculation Line Uncovery (RLU)</td>
<td>9.5</td>
</tr>
<tr>
<td>Lower Plenum Flashing (LPF)</td>
<td>20.</td>
</tr>
<tr>
<td>Feedwater Flashing (FWF)</td>
<td>65.</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>68.</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>93.</td>
</tr>
<tr>
<td>ADS Actuation</td>
<td>128.</td>
</tr>
</tbody>
</table>

(b) PCT related events

<table>
<thead>
<tr>
<th>Events</th>
<th>Time (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Run 942</td>
</tr>
<tr>
<td>Early Dryout at Core Top</td>
<td>2.7</td>
</tr>
<tr>
<td>Rewet at Core Top</td>
<td>partially</td>
</tr>
<tr>
<td>Overall Dryout at Core Top</td>
<td>36.</td>
</tr>
<tr>
<td>Dryout at Core bottom</td>
<td>67.</td>
</tr>
<tr>
<td>PCT Time</td>
<td>22.</td>
</tr>
<tr>
<td>Final Core Quench</td>
<td>197.</td>
</tr>
</tbody>
</table>
| PCT Location*  
  Rod No. | A-88 | A-71 | A-11 |
| Axial Position                | P3 | P4 | P3 |
| PCT Value (K)                 | 901 | 784 | 752 |

* cf. Figs. 2.3 and 2.4

### Table 4.4 Two-phase fluid volume in PV at feedwater flashing initiation time in Run 942

<table>
<thead>
<tr>
<th>Region</th>
<th>Mixture Level (m)</th>
<th>Elevation of Bottom (m)</th>
<th>Mixture Volume (m³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Downcomer</td>
<td>0.90</td>
<td>0.50</td>
<td>0.008</td>
</tr>
<tr>
<td>Guide Tube</td>
<td>1.28</td>
<td>0.29</td>
<td>0.057</td>
</tr>
<tr>
<td>Lower Plenum</td>
<td>0.96</td>
<td>0.0</td>
<td>0.091</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td></td>
<td>0.156</td>
</tr>
</tbody>
</table>
Fig. 4.1 Initial steam quality in average-power bundle.

Fig. 4.2 Initial steam quality in high-power bundle.
Fig. 4.3  Steam dome pressure in three tests.

Fig. 4.4  Feedwater flow rates in three tests.

Fig. 4.5  Downcomer water levels in three tests.
Fig. 4.6 Differential pressure inside core-shroud in three tests.

Fig. 4.7 Typical heater rod surface temperatures of A-11 rod (see Fig. 2.4) at seven elevations in Run 942.

Fig. 4.8 Typical heater rod surface temperatures of A-11 rod (see Fig. 2.4) at seven elevations in Run 943.
Fig. 4.9(a) Dryout and quench times of heater rods in four bundles related with mixture level in Run 942.

Fig. 4.9(b) Dryout and quench times of heater rods in four bundles related with mixture level in Run 943.
Fig. 4.9(c)  Dryout and quench times of heater rods in four bundles related with mixture level in Run 926.

Fig. 4.10  Peak cladding temperatures (PCTs) in three tests (ref. Table 4.3(b)).
5. Break Area Spectrum Test Series

5.1 Introduction

A series of tests were conducted by changing parametrically the break area at the recirculation pump suction line from 0 to 200% of the scaled piping crosssectional area\textsuperscript{1}). The failure of HPCS system was assumed in the tests. The objectives of the tests were:

(1) To study the effect of break size on the scenario of a LOCA
(2) To provide experimental data for assessment of computer codes.

The test results were analyzed by the THYDE-B1 code\textsuperscript{2)} to assess the capability of the code.

5.2 Test Description

The primary test conditions are summarized in Table 5.1. The tests were conducted changing parametrically the break area. The ten tested break areas were 0, 1, 2, 5, 15, 25, 50, 75, 100, and 200% of the scaled (1/424) BWR/6 recirculation loop flow area. A split break at the recirculation pump inlet line was simulated except for one test where a 200% double-ended break was simulated. The HPCS was not used in these experiments to represent the failure of the HPCS diesel generator. ROSA-III tests have shown that the HPCS diesel generator failure results in the highest PCT among possible three single failure modes of the ECCS diesel generators.\textsuperscript{3)}

The initial test conditions were almost the same for all the tests. The steady state power was 3.96 MW yielding MLHR of 16.7 kW/m. The core inlet flow rate was 16 kg/s. The resultant lower plenum subcooling was 11 K and the estimated core outlet quality was 14%.

Test procedures used were standard ones as described in Sec. 2.5. The test procedures for the 0% break test were the same as those for other tests except for no break holes; the main recirculation pumps (MRPs) started coasting down, the core power started decaying following the predetermined curve and feedwater stopped concurrent with the test initiation. This test was conducted as a representative of minimal size break LOCA tests.

5.3 Test Results

5.3.1 LOCA Scenario

The vessel pressures, vessel mixture and collapsed liquid levels and peak power rod cladding surface temperatures measured in the 1% and 100% break tests are shown in Figs. 5.1 and 5.2, respectively. These exemplify the transient parameters during small and large break tests. The mixture levels inside the core shroud and in the lower plenum were estimated from the conduction probe signals. The collapsed liquid level in the downcomer was measured by the differential pressure transducer.

The vessel pressure stayed at 7.4 MPa for a while after the break in the 1% break test and then began to decrease due to core power decay and steam discharge through the steam line until the pressure control system was operated. The system pressure recovered after MSIV closure and was maintained at the setpoint of the simulated safety relief valve of 8.1 MPa. After ADS actuation at 231 s, the pressure began to decrease rapidly.
tion led to injection from LPCS and LPCI.

In the 100% break test, the MSIV was closed before the vessel was depressurized to the pressure control system setpoint (6.7 MPa). The pressure recovery after the MSIV closure was not so prominent as in the 1% break test. A second depressurization began again when the downcomer outlet to the pump suction line was uncovered (recirculation line uncover: RLU) to allow steam discharge through the break. The vessel depressurization induced the LPCS and LPCI actuation. The ADS was tripped on after the LPCS and LPCI actuation and had little influence on the system response in the 100% break.

In the 100% break test the downcomer collapsed liquid level decreased to the bottom, whereas in the 1% break test the downcomer did not become empty throughout the transient as shown in Figs. 5.1 and 5.2.

The lower plenum fluid began to flash when the pressure decreased to 6.4 MPa. The lower plenum flashing occurred during core uncover in the 100% break test. Thus, the core mixture level recovered temporarily. However, in the 1% break test the lower plenum flashing was initiated before the core uncover and had little effect on the core cooling except for the core uncover initiation delay.

Fluid in the feedwater line began to flash at a vessel pressure of 2.2 MPa, almost concurrently with the initiation of LPCS. The inflow from the feedwater line to the pressure vessel due to flashing was detected. The feedwater line flashing slowed the vessel depressurization and thus delayed the initiation of LPCI. The deceleration of the depressurization also resulted in reducing the flashing rate in the two phase mixture in the core and/or lower plenum and thus a decrease in the core or lower plenum mixture level as clearly shown in Figs. 5.1 and 5.2.

The core was entirely uncovered to steam environment in both tests as the system inventory was lost through the break or ADS, or both. Cladding surface temperature excursion propagated from the top of the core downward in both tests as shown in Figs. 5.1 and 5.2 as the core mixture level dropped in both tests.

The core was quenched and reflooded under the effect to LPCS alone, i.e. before the LPCI actuation, in the 1% break test. Cladding temperature turn-around occurred before or upon reflooding. The cladding surface was quenched in a bottom-up fashion at lower core elevations and in a top-down fashion at upper core elevations.

In the 100% break test, the core mixture level recovery and caldding temperature turn-around occurred only after the LPCI actuation. The cladding surface was quenched due to reflooding from the bottom of the core except for the top of the core where the cladding surface was quenched from the top of the core by LPCS.

In the 100% break test, dual mixture levels were formed at the same time in the core and lower plenum after the lower plenum flashing initiation. This indicates the onset of the CCFL phenomena at the core inlet, i.e. limiting of the core liquid drainage to the lower plenum due to updrafting steam flow. This CCFL phenomena suppressed the core mixture level fall and delayed the initiation of the cladding surface temperature excursion.

5.3.2 Pressure Transient

The vessel pressure transients are shown in Fig. 5.3 for all the ten tests. The initial depressurization was arrested by the operation of the pressure control system for break areas $\leq 5\%$ before the MSIV closure. For the break areas $> 5\%$, the MSIV closed before the system pressure decreased to the pressure control systems setpoint of 6.7 MPa. The simulated safety relief valve opened for break area $\leq 5\%$. The steam discharge initiation, which was caused by the ADS actuation for the break areas $< 5\%$, and by the RLU for break the areas $\geq 5\%$, caused
the second and gross vessel depressurization.

5.3.3 Cladding Temperature Transient

The peak cladding temperatures (PCTs) in the ten tests are compared in Fig. 5.4 and in Table 5.2. The PCT occurred at the core mid-plane on the peak power rod upon reflooding caused by LPCI actuation for breaks ≥ 2% except for the 5% break test. For breaks < 2%, the PCTs occurred at Position 3 (353 mm above the core mid-plane) as temperature turn-around occurred under the effect of LPCS alone. The highest PCT was observed in the 50% break test, and was well below the licensing criterion of 1473 K.

The PCT showed correspondence with the timing and duration of the core dryout. The timing of the core dryout mainly depends on the mass loss from the pressure vessel and the timing of core rewetting depends on the vessel depressurization. The major influence of the depressurization was to enable the delivery of ECC water from LPCS and LPCI and thus to terminate the core dryout. In addition, the depressurization caused flashing in various portions of the vessel, which affected the mixture level behavior during core uncoverage.

Timings of the key events for all the ten tests are shown in Fig. 5.5. Three ranges of break area can be defined considering the relative timing and effects of the ADS actuation on the LPCS injection initiation: (a) large break with break areas > 50%, for which the ADS actuation occurred after the vessel had depressurized below the LPCS shut-off head and had only small influence on the vessel pressure transient, (b) intermediate break with break areas between 50% and 5%, for which the ADS actuation occurred after the RLU, but enhanced the vessel depressurization rate, and (c) small break with break areas < 5%, for which the ADS actuation initiated the vessel depressurization and there was no RLU.

It is clear from the figure that the PCT position is uncovered to the steam environment before the actuation of LPCS and LPCI irrespective of a break size. In the large break, PCT position quenching occurred after the LPCI actuation. The rapid depressurization after the RLU induced the actuations of LPCS and LPCI. In the intermediate breaks, the quenching also occurred after the LPCI actuation. The ADS flow area corresponds to 35% of the recirculation line flow area and the ADS actuation becomes more important with the decrease in break area than the RLU for the depressurization to trip on the LPCS actuations. In the small break, PCT position quenching occurred after the LPCI actuation in the 2% break test, however, before the LPCI actuation for break areas < 2%. RLU did not occur in the small break and the vessel depressurization which enabled LPCS and LPCI to initiate injections was caused by ADS actuation.

5.3.4 Core Cooling Effect by ECCS

The measured PCTs are plotted against break area in Fig. 5.6. A dashed line with triangles in the figure shows an adiabatic PCT calculated from cladding surface temperature at the onset of dryout and adiabatic temperature increase by the time of PCT (cladding surface temperature turn-around). The difference between the measured PCT and adiabatic PCT resulted from post-dryout cooling by fluid flow in the core. The effect of the post-dryout cooling will be discussed in the following.

The maximum PCT occurred at a break area of 50%. However, the adiabatic PCT is nearly constant for break areas between 2 and 50%. Thus, the post-dryout core cooling effect was dependent on break area.

The post-dryout core cooling effects were further investigated for two time periods: from the onset of dryout to the LPCS initiation, and from the LPCS initiation to the PCT time. Measured cladding surface temperature increases during these two time periods are
compared with adiabatic temperature increases during these periods in Fig. 5.7. Before the LPCS initiation, the measured and adiabatic temperature increases are roughly proportional. For the period after the LPCS initiation the differences between the measured and adiabatic temperature increases are larger than those for the earlier period. The difference was larger for smaller breaks (< 25%). (Note that the temperature increases after the LPCS initiation were very small for break areas of 0 and 1%). Namely, the effect of LPCS was more prominent for smaller break areas (< 25%). It is evidently shown in Fig. 5.4. The rod surface temperature increasing was suppressed by the LPCS injection initiation for break areas < 25%. The falling-down LPCS water flow rate was limited by the upward steam flow at the upper tie-plate. The upward steam flow rate may primarily depend on the steam discharge rate from the vessel, therefore the ADS and break flow rates. Since the LPCS injection was initiated at the same pressure for all the tests and after the steam discharge initiation, the upper tie-plate steam flow rate during the LPCS injection was smaller for smaller break area. It means that the falling-down LPCS water flow rate increases with decrease in the break area. Thus, the effect of the LPCS water on the rod surface cooling was more significant for smaller break area. Another probable reason why the LPCS core cooling was more effective for smaller break areas is that the smaller the break area, the lower the core power during core dryout, because the core dryout occurred later with decrease in break area. The lower core power resulted in relatively higher ratio of LPCS water core cooling rate to core power generation rate. As a result, the LPCS core cooling became more effective for smaller break areas.

In the 0% and 1% break tests PCTs were detected at Position 3. The LPCS water could easily reach there immediately after the LPCS initiation as shown in Fig. 5.5. The measured and adiabatic temperature increases during the period from the LPCS initiation to the temperature turn-around are very small in the 0% and 1% break tests.

5.4 Analytical Results by THYDE-B1 Code and Discussions

The THYDE-B1 code\textsuperscript{2}) is a fast-running computer code developed for analyzing the thermal-hydraulic response of a BWR during a LOCA. The code is a one-dimensional lumped-parameter code with use of a coarse nodalization as shown in Fig. 5.8.

The major transient parameters obtained from THYDE-B1 code analyses of the 1, 25 and 100% break tests are compared to experimental data in Figs. 5.9, 5.10 and 5.11, respectively. The calculated timings of major events and PCTs are also compared to experimental data in Figs. 5.12 and 5.13, respectively, for all the tests. The vertical axis in Fig. 5.12 is scaled by the normalized core power which is a function of time.

5.4.1 Vessel Pressure

The code successfully predicted the early vessel pressure responses to the operation of the pressure control system, the MSIV closure, and the operation of SRV, because the code calculated well the functions of the main steam line valves and the timings of the valve trip signals based on the downcomer collapsed liquid level.

As discussed in Section 5.3.2, gross vessel depressurization was initiated by the ADS actuation for break areas < 5% and by the RLU for break areas ≥ 5%. The timing of these events, dependent on the downcomer liquid level behaviors, were predicted well by the code.

However, the code overpredicted the peak vessel pressure for break areas ≥ 15%, for which the pressurization after the MSIV closure was arrested by the RLU.

The overprediction of the peak vessel pressure was attributed to the lack of condensation heat transfer model in the THYDE-B1 code. Condensation of vapor on the vessel structure
should have occurred when the vessel was pressurized, and influenced the pressurization rate. Addition of a simple condensation heat transfer model to the code resulted in better agreement between the measured and calculated vessel pressures as typically shown in Fig. 5.10.

After the initiation of gross vessel depressurization, the vessel pressure was primarily dependent on the mass and energy release rates from the vessel, until significant amount of steam was generated following the feedwater line flashing and core reflooding. The vessel depressurization behavior was predicted well by the code which uses the Moody critical flow model with a discharge coefficient of 0.6 for the calculation of break and ADS flow rates.

The later vessel depressurization behavior determined the timings of ECC water injection from LPCS and LPCI, the only ECCSSs available in the present tests. The timing of the LPCI injection, with a pump shutoff head of 1.58 MPa, was affected significantly by the flashing in the feedwater line, which began at a pressure of 2.2 MPa and decreased the vessel depressurization rate thereafter. Thus, addition of a feedwater line node to the model shown in Fig. 5.8 improved the prediction of the vessel depressurization behavior as shown Fig. 5.10. However, with such model modification the predicted vessel depressurization after the onset of feedwater line flashing was still faster than the data, probably because additional vapor generation associated with the feedwater line flashing was not considered. Namely, the feedwater line flashing caused vaporization of saturated liquid and expelled the liquid from the feedwater line into the vessel, and the liquid came in contact with the heated vessel structures and internals. The code cannot calculate such evaporation of liquid in the vessel, because its three-region node model assumes that liquid injected into the vapor region immediately enters into the mixture region after mixing with vapor.

5.4.2 Core Mixture Level

Prediction of the core mixture level is important because the rod temperature closely responds to the mixture level behavior, particularly during the core uncovering period. The factors determining the core uncovering process are: the vapor separation from the mixture, vapor generation due to flashing and heat transfer, and flow through the jet pump discharge line connecting the lower plenum and downcomer.

Core uncovering initiated in the present tests only after the onset of gross vessel depressurization for all the break area. The entire core was uncovered for break areas $\geq 1\%$. The same results were obtained from the calculations.

The core uncovering process was affected by flashing of liquid inside the core shroud. The core mixture level was raised temporarily by the LPF initiated at a pressure of 6.4 MPa, and by the actuation of the ADS, both increased vapor generation in the core mixture. In the tests the LPF occurred during the core uncovering process, for break areas $\geq 5\%$, and during the ADS actuation for break areas $< 15\%$. Following the LPF, vapor-full regions were formed below the core inlet for break areas $\geq 5\%$, first beneath the lower tie-plate, and then beneath the core side entry orifice, while there still existed two-phase mixture in the core. Thus, the onset of the CCFL phenomenon at the core inlet flow area contractions was evident. The code of course did not predict this phenomenon because the CCFL phenomenon is not modeled in the code. Furthermore the code assumes a uniform distribution of voids in the mixture.

The core mixture level swell following the LPF was generally underpredicted. Here, it is important to note that the LPF was a gradual process in the experiments, because there was an uneven temperature distribution in the lower plenum subcooled liquid. The conductivity probe signals and temperature measurements show that flashing first occurred at the core inlet, between the side-entry orifice and the lower tie plate, before the whole lower plenum fluid began to flash. Thus, there was interaction between the vapor and liquid phases at the
flow area contractions before the gross flashing in the lower plenum began. Such interactions may well have reduced liquid drainage from the core before the onset of obvious CCFL and thus have raised the core mixture level.

On the other hand, the code assumes a uniform temperature distribution in the subcooled region, so that flashing occurs suddenly when the saturation pressure corresponding to the temperature is reached. Thus, precursory flashing at the core inlet and the upper portion of the lower plenum was not calculated. Such underprediction of the flashing rate before initiation of gross LPF, together with neglect of the interaction between phases at the core inlet, may well have been responsible for the underprediction of the core mixture level during the typically early core recovery process shown in Fig. 5.10.

The underprediction of core mixture level had the most significant influence on analytical results for the 15 and 25% break tests. In the analyses of these tests, the recovery of Position 3 occurred before the onset of LPF, whereas in the actual tests, the recovery of Position 3 occurred after the onset of gross LPF for all the break areas. Such a difference in the mixture level behavior affected the PCT prediction for these tests.

5.4.3 Rod Surface Temperature

Rod surface temperature excursions occurred as the core was uncovered, and continued until the rod surface heat transfer was improved due to either the actuation of core spray (LPCS) or core reflooding caused by the LPC1. Thus, the timing of these events and the post-dryout heat transfer coefficients are important for prediction of rod temperature behaviors. The timing of dryout is determined by the core mixture level behavior, and the timings of ECC water injection are determined by the vessel pressure behavior.

The post-dryout heat transfer coefficients were code input for the present analyses. Empirical values obtained from ROSA-III tests were used. These are 40 W/m²K for the period before the initiation of LPCS, and 70 W/m²K for the period thereafter until refloodings. Because these heat transfer coefficients were determined for the rod surface superheating above the fluid saturation temperature, they are applicable to the THYDE-B1 code which assumes the vapor phase to be always saturated. These heat transfer coefficients are considerably greater than the 10CFR50 Appendix K convection and spray heat transfer coefficients of 0 and 7.3 to 17.1 W/m²K (1.5 to 3.5 BTU/ft²h°F) respectively. The radiation heat transfer was neglected. The calculated temperature behavior was in satisfactory agreement with the experimental data when the timing of dryout was calculated correctly as shown in Figs. 5.9 through 5.11. The effect of LPCS was well calculated; the initiation of LPCS led to temperature turn-around at the upper half of the core including the midplane (Positions 1 through 4) for break areas ≤ 25% in the tests, and for break areas ≤ 50% in the calculations. Thus, the timing of core reflooding, caused by the LPC1, influenced the PCT only for the larger break areas (see Fig. 5.12).

5.4.4 Dependence of PCT on Break Area

The measured and calculated PCTs are compared in Fig. 5.13. The measured PCTs occurred at Position 3 for break areas ≤ 2%, and at Position 4 (core midplane) for the larger break area. Figure 5.13 also shows the ranges of peak temperature measured at Positions 3 and 4 on peak-power rods. (Eleven peak-power rods out of twenty in total were instrumented.) Considerable rod-to-rod variation of the peak temperature, ranging from 10 to 80 K, was observed at both Position 3 and 4 elevations. In the following discussion the calculated peak temperatures are compared to the measured maximum-peak temperatures.

The calculated PCTs occurred at Position 3 for all the break areas, and were higher than
the measured PCTs. Both the Position 3 and 4 peak temperatures were overpredicted, except for the break area of 25% for which the Position 4 peak temperature was underpredicted (but higher than the measured minimum-peak temperature at the same elevation). Because of the considerable overprediction of the Position 3 peak temperature for the break area of 25%, the maximum calculated PCT of 1015 K was obtained for this break area, whereas the maximum measured PCT (930 K) was obtained for the break area of 50%. The overprediction of the Position 3 peak temperature for the break area of 25% (and 15%) resulted from earlier uncover in the calculation than in the test as has been discussed and shown in Fig. 5.12. In the THYDE-B1 calculation Position 4 was temporarily rewetted after the ADS actuation for the break area of 25% as shown in Fig. 5.10, which resulted in the low peak temperature for a 25% break at Position 4.

In summary, the present analyses revealed the capability of the THYDE-B1 code, designed for prediction of slow transients associated with small breaks, to predict fairly well the overall thermal-hydraulic behavior for large break areas ranging between 50 and 200% as well. Calculated chronologies of major events for all tests are summarized in Fig. 5.12 in comparison with measured ones. The THYDE-B1 code predicted fairly well these for entire test range from large break to small break. The present analyses also revealed the sensitivity of the predicted thermal-hydraulic responses during intermediate-break tests to modeling of the vessel structure heat transfer and the spatical distributions of void fraction.

5.5 Conclusions

From the recirculation-line break tests conducted for a spectrum of break areas and a failed HPCS, the following conclusions were drawn:

1. Core uncover occurred for all the break areas as the vessel mass inventory was depleted through the break, or ADS, or both. The entire core was uncovered for break areas ≥ 1%.

2. Steam discharge depressurized the vessel to the low pressure ECCS (LPCS and LPCI) injection pressures, and thus enabled safe recovery of core cooling without HPCS. The major steam leak paths were ADS for break areas < 5%, the break for break areas > 50%, and both for the intermediate break areas.

3. Flashing in the lower plenum and in the feedwater line temporarily raised and reduced the core mixture level, respectively, when they occurred during the core uncover period.

4. The rod surface dryout led to temperature excursion. Temperature turn-around occurred as rod surface cooling was improved due to the core top spray (LPCS) for break areas < 2% or bottom-up reflooding caused by the LPCI for break areas ≥ 2%. The effect of the LPCS was more pronounced for smaller break areas especially at core upper-elevations. The PCT was limited by the LPCS for break areas ≤ 25%, and by LPCI for the larger break areas.

5. The measured PCTs ranged from 640 to 930 K and were much lower than the present licensing criterion of 1473 K.

From the experimental analyses with the THYDE-B1 code the following conclusions have been drawn:

1. The core mixture level behavior was predicted fairly well by the code using a huge-node model (representing the total vessel volume by two nodes), however, uncover of the core upper-portions was calculated to occur earlier than the tests.

2. The rod temperature behavior was predicted fairly well when empirical post-dryout heat transfer coefficients obtained from ROSA-III tests were used.
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Fig. 5.4 Rod surface temperature behavior at the location where PCT occurred - comparison of data from all the tests
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Fig. 5.13 Comparison of measured and calculated peak cladding temperatures
### Table 5.1 Test conditions

<table>
<thead>
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<th>Break Conditions</th>
<th>[0, 1, 2, 5, 15, 25, 50, 75, 100, 200%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Position</td>
<td>Recirculation Pump Inlet</td>
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<tr>
<td>Area</td>
<td>[0, 1, 2, 5, 15, 25, 50, 75, 100, 200%]</td>
</tr>
<tr>
<td>ECCS Conditions</td>
<td>HPCS Failure</td>
</tr>
<tr>
<td>Steady State Conditions</td>
<td></td>
</tr>
<tr>
<td>Steam Dome Pressure</td>
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</tr>
<tr>
<td>Lower Plenum Subcooling</td>
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<td>Core Exit Quality</td>
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### Table 5.2 Comparison of peak cladding temperatures

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<tr>
<th>Break Area (%)</th>
<th>PCT (K)</th>
<th>Time (s)</th>
<th>Position</th>
<th>Rod&lt;sup&gt;c&lt;/sup&gt;</th>
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<tr>
<td>0</td>
<td>637</td>
<td>696</td>
<td>P3&lt;sup&gt;a&lt;/sup&gt;</td>
<td>A11</td>
</tr>
<tr>
<td>1</td>
<td>754</td>
<td>546</td>
<td>P3</td>
<td>A87</td>
</tr>
<tr>
<td>2</td>
<td>804</td>
<td>531</td>
<td>P4&lt;sup&gt;b&lt;/sup&gt;</td>
<td>A68</td>
</tr>
<tr>
<td>5</td>
<td>835</td>
<td>410</td>
<td>P4</td>
<td>A17</td>
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<td>P4</td>
<td>A71</td>
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<tr>
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<td>189</td>
<td>P4</td>
<td>A82</td>
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<tr>
<td>75</td>
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<td>785</td>
<td>119</td>
<td>P4</td>
<td>A71</td>
</tr>
</tbody>
</table>

<sup>a</sup> 353 mm above the mid-plane of core.

<sup>b</sup> Mid-plane of core.

<sup>c</sup> All rods are peak power rods in the peak power bundle. (see Fig. 2.4(b)).
Fig. 5.1 Vessel pressure, vessel mixture/collapsed level and peak-power rod surface temperature transients in 1% break test.
Fig. 5.2 Vessel pressure, vessel mixture/collapsed level and peak-power rod surface temperature transients in 100% break test.
Fig. 5.3 Vessel pressure transients - comparison of data from all the tests.

Fig. 5.4 Rod surface temperature behavior at the location where PCT occurred - comparison of data from all the tests.
Fig. 5.5 Chronology of major events - comparison of data from all the tests.

Fig. 5.6 Measured peak cladding temperatures (PCT) versus PCTs calculated neglecting the post-dryout core cooling effects.
Fig. 5.7 Rod surface temperature increases during the post-dryout time periods before and after LPCS actuation - comparison between experimental data and adiabatic calculation.

Fig. 5.8 THYDE-B1 nodalization for ROSA-III split-break test configuration.
Fig. 5.9  Comparison of measured and calculated major transient parameters - 1% break test.
Fig. 5.10  Comparison of measured and calculated major transient parameters - 25% break test.
Fig. 5.11  Comparison of measured and calculated major transient parameters - 100% break test.
**Fig. 5.12** Comparison of measured and calculated chronologies of major events.

**Fig. 5.13** Comparison of measured and calculated peak cladding temperatures.
6. International Standard Problem No.12 (ISP-12)

6.1 Introduction

Run 912\textsuperscript{1)} of ROSA-III experimental program was selected as the OECD/NEA CSNI (Committee on Safety of Nuclear Installations) international standard problem number twelve (ISP-12)\textsuperscript{2)}, which was the first standard problem concerning a BWR LOCA. Run 912 was conducted on May 19, 1981 simulating a LOCA with the 5% split break at the pump suction in the recirculation line piping. The HPCS was assumed to fail. One of the important activities of the CSNI LOCA/ECC and Fuel Behavior Working Group is the international standard problem (ISP) exercise, in which participants from member countries analyze the data of the selected test using different computer codes and/or calculation models and calculated results are compared with test data in order to accumulate the knowledge to use the codes and also to improve them.

Eight participants submitted the calculations of Run 912 transients by April 20th, 1982. Table 6.1 lists the participants names and computer codes used in the analysis.

Comparisons of calculated and test results for important variables are presented in this chapter. The two-dimensional effects on core thermal-hydraulics in a small break LOCA\textsuperscript{3)} are also discussed in this chapter through the experimental analysis of the Run 912 test data.

6.2 Test Conditions and Test Procedure

The initial fluid conditions of Run 912 before the break initiation were as follows. The steam dome pressure was 7.35 MPa. The fluid temperature in the lower plenum was 551.8 K with the subcooling of 10.5 K. The electric power supplied to the core was 3.97 MW corresponding to 44% core power if the power/volume ratio were the same as in a BWR core. The maximum linear heat generation rate of the simulated fuel rods in the core was 16.7 kW/m. The core inlet flow rate was 16.0 kg/s and the estimated steam quality at the outlet of the core was 13.5% assuming that 5% of the coolant flowing into the lower plenum bypassed the core.

The test was initiated by opening a quick opening blowdown valve located downstream the orifice which simulated the break size of 5% of scaled (1/424) recirculation line pipe cross section. The main recirculation pumps were tripped at the break initiation.

The core power was not changed in the first 8.8 s after the break initiation and was decreased afterwards in accordance with the decay curve of the stored heat release and the heat generation rate after the reactor scram (see Fig. 2.5). The radial power and axial power distributions in the core shown in Figs. 2.3 and 2.4 were the same in the transient and in the steady state.

Closure of the feedwater line was initiated at 2.0 s and completed at 3.1 s after the break initiation. The MSIV closure was initiated at 22 s and completed at 24 s after the break initiation. The CV-130 (see Fig. 2.2) was opened between 80 s and 110 s after the break initiation simulating the safety relief valve actions after the MSIV closure. The ADS was actuated at 158 s after break. The system pressure decreased sharply by actuation of the ADS and the LPCS started to spray water from the top of the core at a system pressure of 2.38 MPa at 318 s after the break initiation. The LPCI started to inject water into the core bypass at a system pressure of 1.81 MPa at 406 s after the break initiation reflooding the core shortly after
actuation.

6.3 Comparison of Calculated Results with Test Data

Eight participants submitted calculations for ISP-12. Table 6.1 lists the participants and computer codes used. USA used TRAC-BD1 and Sweden and JAERI-R used RELAP5/MOD1. These two codes are so-called advanced codes based on non-homogeneous nonequilibrium models. JAERI-C used the THYDE-B1 code which was developed at JAERI for BWR small break LOCA analysis and has three-region node model with moving boundaries used also in the SAFE code. The other participants used RELAP4/MOD6 or RELAP4/MOD6/U4/J3. The latter is an improved version of RELAP4/MOD6 revised at JAERI including improvements for analysis of refill and reflood phases of not only large break but also small break LOCAs in a BWR.

6.3.1 System Pressure

Comparison of system pressure calculations and data is shown in Fig. 6.1. The trends of the system pressure were calculated well by most participants. A considerably higher pressure than the test data was calculated by Switzerland and Netherlands before ADS actuation without using safety relief valve model in RELAP4/MOD6. The system pressure after ADS actuation was considerably lower in the Sweden and JAERI-R calculations than the measured pressure and higher in the JAERI-A calculation primarily due to insufficient accuracy in the calculation of the ADS flow.

6.3.2 Break Flow

Break flow measurement using three drag disks and a gamma densitometer involved an error of at least ±20%. Accordingly emphasis was made on calculation of the incipient time of decrease in break flow due to uncovering of the recirculation line inlet (RLU) in the downcomer. Comparison of calculated and measured break flows is shown in Fig. 6.2. The uncovering time of the recirculation line inlet calculated by most participants agreed with the measured time of 160 s within +10 s to -25 s. The break flow calculated by USA was much higher than the other results and the uncovering time of the recirculation line inlet was 50 s earlier than the test result since the calculated break flow seemed to be adjusted as a boundary condition to the data which had large uncertainties. The uncovering time of the recirculation line inlet in the Switzerland calculation was also earlier than the data, while the break flow rate was smaller than the other calculation results except for the JINS result. This was probably due to smaller initial mass inventory in the downcomer in the analysis.

6.3.3 Total Core Inlet Flow

Comparison of calculated and measured total core inlet flows is presented in Fig. 6.3. The total core inlet flow decreases to a very low value at 50 s after the break while there remains still enough water in the pressure vessel for core cooling. Models for the recirculation pump and the jet pump, therefore, are not so important in a small break LOCA as in a large break LOCA. The core cooling, however, can be underestimated within 50 s after the break if the calculated total core inlet flow becomes considerably lower than the measured flow. The trend of the decrease in the total core inlet flow until 50 s after the break was calculated well by most participants. Quantitatively higher flow rate than the data was calculated by JINS, JAERI-C, JAERI-A and JAERI-R, who used incomplete pump data for the ROSA-III recirculation pump without measured torque data. The other four participants seem to have
used the experimental pump coastdown curve (pump speed vs. time) to calculate the core inlet flow rate accurately. The calculated flow rate, however, was considerably smaller than the measured flow rate for a certain time period within 50 s after the break in the USA, Switzerland and Sweden calculations. This decrease in the total core inlet flow rate brought about temporary dryout at the upper part of the core in the three calculations.

6.3.4 Mixture Level

Comparison of calculated and measured mixture levels in the average power channel in the core is presented in Fig. 6.4. Since advanced codes like TRAC-BD1 and RELAP5/MOD1 do not predict the mixture level transient, in the present comparison the mixture level for the advanced codes was defined as the level separating the regions with void fractions of greater and less than 0.95.

In the USA and Switzerland calculations, the upper part of the core was uncovered immediately after break. This was probably due to a temporary decrease in the total core inlet flow rate immediately after the break.

In the USA calculation, the mixture level decrease in the core was initiated considerably earlier than the data probably due to the larger break flow rate than the test, and the mixture level reached below the midplane of the core. The mixture level increase due to lower plenum flashing (LPF) was slower than the data, and mixture level decrease after LPF was also slower. The minimum level after LPF in the USA calculation was at the midplane of the core, whereas the whole core was uncovered in the test.

The Switzerland calculation gave earlier initiation of the mixture level decrease in the core, approximately 55 s earlier than the data. This was probably due to the small initial inventory of the coolant assumed in the calculation. Recovery of the mixture level due to LPF was smaller in the calculation than in the data. The mixture level decrease after LPF was slower in the calculation than in the data, and the whole core uncovered did not occur in the calculation.

The JAERI-A calculation gave a 16 s delay for initiation of LPF. The calculation also gave a slower decrease of the mixture level in the core after LPF.

The other calculations by JINS, JAERI-C and JAERI-R gave good agreements with the data on the mixture level transient in the core.

6.3.5 Heater Rod Surface Temperature

There is a strong correlation between the heater rod surface temperature transient and the mixture level transient in the core. Temperature rise or dryout corresponds to exposure of the heater rod surface to steam above the mixture level, and the temperature turnaround or decrease corresponds to covering of the surface by two-phase mixture below the mixture level. Comparison of calculated and measured heater rod surface temperatures at three elevations in the core, i.e., top part, midplane and bottom part, is presented in Fig. 6.5.

Temporary dryout and rewet within 40 s after the break were calculated at the top part of the average power bundle in the USA and Switzerland calculations, while they did not occur in the test. This is because the liquid level in the core decreased temporarily by the decrease in the core inlet flow rate below the test data. The trends of the heater rod surface temperatures calculated by USA and Switzerland were in poor agreement with the measured result because of poor predictions of the mixture level transient in the core. No rewetting was calculated at the top part of the core after LPF and no dryout was calculated at the bottom part of the core after mitigation of LPF.

The trends of dryout and rewetting were calculated well by RELAP5/MOD1 (Sweden,
JAERI-R), RELAP4/MOD6/U4/J3 (JINS, JAERI-A) and THYDE-B1 (JAERI-C). A considerable delay in the dryout in the JAERI-A calculation, however, was probably due to inappropriate selection of the parameters for the mixture level calculation. The temperature increasing rates after dryout in the JINS and JAERI-A calculations were larger than the experimental value since the evaluation-model heat transfer correlations were used in RELAP4/ MOD6/U4/J3 after initiation of LPCS. Top-down quenching was calculated by RELAP4/ MOD6/U4/J3 (JINS, JAERI-A). However, the heater rod surface temperature started to rise again after reflooding because of a conservative heat transfer coefficient used in the post reflooding phase in these analyses.

The temperature increasing rates after dryout calculated by RELAP5/MOD1 (cycle 001) (JAERI-R) and THYDE-B1 (JAERI-C) were lower than the measured increasing rate. In the THYDE-B1 analysis the shroud inside region is expressed by one node, therefore, no super-heated steam condition can be calculated, resulting in the lower heater rod surface temperature after dryout. The RELAP5/MOD1 (cycle 001) code calculated a high heat transfer coefficient by a heat transfer correlation for “two-phase laminar natural convection” after dryout. This high heat transfer coefficient produces the lower temperature increasing rate than the data after dryout. This heat transfer correlation has been modified in the RELAP5/MOD1 (cycle 014) code.

6.4 Two-Dimensional Core Thermal-Hydraulic Phenomena

6.4.1 Mixture Level

The two-phase mixture levels in the pressure vessel were measured by conductivity probes. The measured mixture levels in the upper plenum, the core, the core inlet channel, the lower plenum, the guide tube and the downcomer are presented in Fig. 6.6. The mixture levels in the core and the core inlet channel were measured in the four channels, and the level in the upper plenum was measured above the channels A and C.

There was no significant difference in the mixture level transients among the four channels. While the top part of the core in the peak power channel A was uncovered slightly later than in the average power channels B, C and D, the bottom parts of the four channels were uncovered almost at the same time. The reflooding started at about the same time in the four channels, whereas the top part of the peak power channel was reflooded slightly earlier than the average power channels.

6.4.2 Dryout and Quenching

The timing of the dryout after lower plenum flashing at the same elevation was the same for all the fuel rods in the peak power bundle A, independent of the local peaking factor. This is because the dryout occurred due to exposure of the fuel rod surface to steam above the mixture level. The same statements are true for all the fuel rods in the average power bundles B, C and D. The increasing rate of the surface temperature after the dryout at the same position in the peak power bundle was similar for all the fuel rods with the same local peaking factor. The same statement is true for all the fuel rods in the average power bundles. The increasing rate of the surface temperature depended on the linear heat rate of the fuel rod (i.e., a product of the axial peaking factor, the radial peaking factor and the local peaking factor). There were no significant differences in the timing of dryout and the increasing rate of surface temperature after the dryout among the average power bundles B, C and D.

The fuel rods which faces the channel box wall and the other fuel rods are termed “peripheral fuel rods” and “central fuel rods”, respectively, in this chapter. Dryout and
quenching fronts were estimated from the surface temperatures of the fuel rods. The dryout and quenching fronts of the peripheral fuel rods and the central fuel rods in the peak power bundle A are presented in Fig. 6.7(a) and (b), respectively. The mixture level transient in the peak power channel A is also shown in Fig. 6.7(a) and (b).

All the dryout fronts after lower plenum flashing in the peak power bundle were almost the same and agreed with the mixture level transients in the peak power channel. This is because the dryout occurred due to exposure of the fuel rod surface to steam above the mixture level, and the mixture level transient was uniform in the peak power channel as shown in Fig. 6.6. The same statements are true for the dryout fronts in the average power bundles.

6.4.3 Quenching Pattern, PCT and Quenching Time

In this section, the fuel rods are divided into the three groups according to the quenching pattern:

- **Pattern I** — fuel rods which showed temporary rewetting at the upper part of the core due to CCFL break-down before LPCS actuation, and top-down and bottom-up quenching after LPCS actuation

- **Pattern II** — fuel rods which showed top-down and bottom-up quenching after LPCS actuation

- **Pattern III** — fuel rods which showed only bottom-up quenching after LPCS actuation.

Fig. 6.8 shows the peak cladding temperatures (PCTs) at Position 4, the quenching patterns and the quenching times of Position 4 for all the instrumented rods. The PCT is defined for each fuel rod in this chapter as the highest cladding temperature during excursion. The quenching patterns only for the fuel rods which have all seven thermocouples are presented in Fig. 6.8.

The number and the percentage of fuel rods of each quenching pattern are summarized in Table 6.2, dividing into sixteen categories according to the bundle power (i.e. the radial peaking factor), the local peaking factor and the rod location (i.e. the peripheral fuel rod or the central fuel rod). The percentage of pattern I fuel rods in the peripheral fuel rods was larger than in the central fuel rods, and that of pattern III fuel rods in the central fuel rods was larger than in the peripheral fuel rods. The steam generated in the core or the lower plenum flowed upward mainly in the central region of the channel, which limited water fall-back from the upper plenum to the core. This led to early CCFL break-down at the upper tieplate in the peripheral region, and the water falling from the upper plenum due to CCFL break-down dropped down mainly through the peripheral region in the channel. The percentage of pattern I fuel rods in the average power bundles B, C and D was larger than in the peak power bundle A, and that of pattern III fuel rods in the peak power bundle was larger than in the average power bundles. The reason may be that the dry surface rewetted and was quenched more easily by falling water in the average power bundles than in the peak power bundle since the fuel rod surface temperatures in the average power bundles were lower than in the peak power bundle.

The average PCT and average quenching time at Position 4 of the fuel rods in each quenching pattern are presented in Table 6.3. The PCT in each category became higher in the order, pattern I, II, III, and the quenching time became later in this order. The temporary rewetting by CCFL break-down in the pattern I fuel rods and the top-down quenching in the pattern I and II fuel rods had effect on the PCT and the quenching time. The PCT increased with the linear heat rate of the fuel rod i.e. the product of the radial peaking factor and the local peaking factor. When the local peaking factor was the same, the average PCT in the central fuel rods was higher than that in the peripheral fuel rods in both the peak power
bundle and the average power bundles. This is because the percentage of pattern I fuel rods in the peripheral fuel rods was larger than in the central fuel rods, and that of pattern III fuel rods in the central fuel rods was larger than in the peripheral fuel rods. Since the high power fuel rods were located in the peripheral region, the temporary rewetting due to CCFL break-down and the top-down quenching tended to decrease the PCTs of the high power rods in the peripheral region.

The quenching time in each category became later in the order, pattern I, II, III. Generally, the quenching time became later as the local peaking factor increased when the rod location was the same. The quenching times averaged over pattern I, II and III rods in the peak power bundle were almost the same regardless of the local peaking factor and the rod location (i.e. the peripheral fuel rod or the central fuel rod). The reason may be that the pattern I fuel rods occupied fairly large part of the peripheral fuel rods and the pattern III fuel rods occupied fairly large part of the central fuel rods, and the fuel rods with a local peaking factor of 1.1 were located only in the peripheral region and those with a local peaking factor of 0.875 were located only in the central region. The same statements are true for the quenching times in the average power bundles. The quenching time in the peak power bundle was later than that in the average power bundles. Since the fuel rod surface temperatures in the average power bundles were smaller than in the peak power bundle, the dry surface rewetted and was quenched more easily by falling water resulted from CCFL break-down, or LPCS actuation, in the average power bundles than in the peak power bundle.

There were no significant differences in the PCT and the quenching time among the average power bundles B, C and D.

6.5 Conclusions

6.5.1 Conclusions obtained in ISP-12
The following conclusions were obtained from comparisons between test data and eight participants calculations using five different computer codes.

(1) The trends of the system pressure were calculated well by most participants. However, a considerably higher pressure before ADS actuation without using the safety relief valve model in RELAP4/MOD6 was calculated by two participants.

(2) The trends of dryout and rewetting were calculated well by RELAP5/MOD1, RELAP4/MOD6/U4/J3 and THYDE-B1. However, a considerable delay in the dryout was calculated in one of the RELAP4/MOD6/U4/J3 analysis (JAERI-A) because of inappropriate selection of parameters for the mixture level calculation. The temperature increase rate after dryout was larger than the experimental value in the RELAP4/MOD6/U4/J3 analyses because the evaluation model heat transfer correlations were used in RELAP4/MOD6/U4/J3 after initiation of LPCS. The temperature increase rate after dryout calculated by RELAP5/MOD1 and THYDE-B1 was lower than the measured increase rate. In the THYDE-B1 analysis the shroud inside region is expressed by one node, therefore, no super-heated steam condition can be calculated, resulting in the lower heater rod surface temperature after dryout. The RELAP5/MOD1 (cycle 001) code used the heat transfer correlation for the laminar natural convection after dryout. The applicability of this heat transfer mode to the region after dryout should be assessed.

6.5.2 Conclusions about Two-Dimensional Core Thermal-Hydraulic Phenomena
The following conclusions have been obtained from the test results on the two-dimensional effects for core thermal-hydraulics in a small break LOCA.
(1) There is no significant difference in the mixture level transient among the peak power channel and the three average power channels.

(2) The timing of the dryout of a fuel rod does not depend on the local peaking factor. The increasing rate of the surface temperature after the dryout depends on the peaking factor or the local linear heat generation rate of a fuel rod.

(3) The peak cladding temperature and quenching time of a fuel rod depend on the linear heat rate of a fuel rod and its location in the bundle (i.e. the peripheral fuel rod or the central fuel rod). The peak cladding temperature of a fuel rod increases with the linear heat rate.

(4) The peak cladding temperatures of high power fuel rods were limited to lower values since they were located in the peripheral region where temporary rewetting before LPCS actuation occurred often.

(5) There is no significant differences in trends of the peak cladding temperature and the quenching time among the three average power bundles.

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   (b) Central fuel rods
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<table>
<thead>
<tr>
<th>Country (Organization)</th>
<th>Identification</th>
<th>Computer Code</th>
<th>Time Interval (s)</th>
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<td>U</td>
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*1 Idaho National Engineering Laboratory  
*2 Swiss Federal Institute for Reactor Research  
*3 Studsvik Energiteknik Ab, Technical Research Center of Finland  
*4 Netherlands Energy Research Foundation  
*5 Institute of Nuclear Safety, Japan  
*6 Japan Atomic Energy Research Institute, Division of Nuclear Safety Evaluation Nuclear Safety Code Development Laboratory  
*7 Japan Atomic Energy Research Institute, Division of Nuclear Safety Evaluation Nuclear Safety Analysis Laboratory  
*8 Japan Atomic Energy Research Institute, Division of Nuclear Safety Research Reactor Safety Laboratory I
### Table 6.2 Number and percentage of fuel rods in each quenching pattern

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<th>Location</th>
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<td>1.0</td>
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<td></td>
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<td>C</td>
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<td>1.0</td>
<td>P</td>
<td>0</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C</td>
<td>4</td>
<td>1 (25%)</td>
<td>2 (50%)</td>
<td>1 (25%)</td>
</tr>
<tr>
<td></td>
<td>0.875</td>
<td>P</td>
<td>0</td>
<td>-</td>
<td>-</td>
<td>-</td>
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<tr>
<td></td>
<td></td>
<td>C</td>
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<td>10 (67%)</td>
<td>5 (33%)</td>
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<tr>
<td>B, C, D (1.0)</td>
<td>1.1-0.875</td>
<td>P</td>
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<tr>
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<td>C</td>
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<td>2 (40%)</td>
<td>2 (40%)</td>
<td>1 (20%)</td>
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- RPF: Radial peaking factor
- LPF: Local peaking factor
- P: Peripheral fuel rod, C: Central fuel rod
- (a) (b) (c) (d) correspond to the notes at the bottom of the table.

### Table 6.3 Average PCT and quenching time of Position 4 in each quenching pattern

| Bundle (RPF) | LPF | Location | Average PCT (Average quenching time)
<table>
<thead>
<tr>
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<td>----------</td>
<td>--------------------------------------</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Pattern I</td>
<td>Pattern II</td>
</tr>
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<td>P</td>
<td>((2))781(397)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C</td>
<td>((0)) -</td>
</tr>
<tr>
<td></td>
<td>1.0</td>
<td>P</td>
<td>((2))755(369)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C</td>
<td>((0)) -</td>
</tr>
<tr>
<td></td>
<td>0.875</td>
<td>P</td>
<td>((0)) -</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C</td>
<td>((4))737(384)</td>
</tr>
<tr>
<td>B, C, D (1.0)</td>
<td>1.1</td>
<td>P</td>
<td>((2))716(367)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C</td>
<td>((0)) -</td>
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<td>C</td>
<td>((1))709(360)</td>
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<td></td>
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<td>P</td>
<td>((0)) -</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C</td>
<td>((0)) -</td>
</tr>
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</table>

- RPF: Radial peaking factor
- LPF: Local peaking factor
- P: Peripheral fuel rod, C: Central fuel rod
- (a) (b) (c) (d) correspond to the notes at the bottom of the table.
Fig. 6.1 Comparison of system pressures.

Fig. 6.2 Comparison of break flows.
Fig. 6.3 Comparison of total core inlet flows.

Fig. 6.4 Comparison of mixture levels in the core.
Fig. 6.5 Comparison of heater rod surface temperatures.
(c) Position 7 (50 mm above the bottom of the core)

Fig. 6.5 Comparison of heater rod surface temperatures.

Fig. 6.6 Mixture levels in upper plenum, core, core inlet channel, lower plenum, guide tube and downcomer.
Fig. 6.7 Dryout and quenching fronts in peak power bundle A.
Fig. 6.8  PCT at Position 4, quenching pattern and quenching time of Position 4.
7. Effectiveness of Automatic Depressurization System on Core Cooling

7.1 Introduction

The ADS is a part of the ECCS of a BWR. The ADS is actuated to depressurize the primary cooling system in a SBLOCA to operate the low pressure ECCS, i.e., the LPCS system and the LPCI system to adequately cool the core even if the HPCS system fail to start. The ADS is designed to actuate by the low water level (L1) in the reactor vessel or the high pressure in the drywell with the time delay of 120 s.

In a LBLOCA, the ADS plays little role because the LPCS and LPCI are actuated due to the rapid depressurization by the break flow itself. The ADS ability of depressurization becomes significant in a SBLOCA to actuate the LPCS and LPCI. However, the ADS can be considered as another break with a break area of 34% of the recirculation line cross section area and the mass discharge through ADS becomes a determinant factor to uncover the core in a SBLOCA with a very small break area. Therefore, the ADS has positive and negative effects on the core cooling in a BWR SBLOCA and it is important to examine the effect of ADS in a SBLOCA.

The ROSA-III SBLOCA tests were performed to investigate the primary thermal-hydraulic behavior with the special interest in the ADS effect on the core cooling under the different ADS conditions of actuation time and flow area. The LPCS single failure was assumed in the experiments because the HPCS single failure is the severest single failure assumption of ECCS for core cooling.

The analyses of the ROSA-III tests and the BWR LOCA were also performed with THYDE-B1/MOD2 code to check the code predictability and investigate effects of the ADS on the thermalhydraulic behavior during the BWR SBLOCA.

7.2 Test Conditions

ROSA-III test conditions are summarized in Table 7.1 for the four small break tests with different ADS conditions. In the reference tests, Runs 922 and 921, the ADS was actuated by the L1 level signal with a time delay of 120 s (L1 + 120 s) with the rated flow area as designed. Two 5% break tests, Runs 922 and 932 with the ADS flow areas of 100 and 50%, respectively, were performed to investigate the effect of the ADS flow area on the thermalhydraulic behavior. Two 1% break tests, Runs 921 and 931 with the ADS actuation timing of L1 + 120 s and L2 + 120 s, respectively, were performed to investigate the effect of the ADS actuation timing on the thermalhydraulic behavior.

The primary initial steady state test conditions were common to these four tests. The steam dome pressure was 7.35 MPa and the lower plenum subcooling was 10.5 K. The core power was 3.96 MW corresponding to 44% of the scaled steady power of 9 MW. The core inlet flow rate was 16 kg/s and the core outlet quality in the upper plenum was estimated to be 12%. The scram level L3 was specified as the initial downcomer liquid level.
7.3 Test Results

7.3.1 Effect of ADS Flow Area

The effect of ADS flow area was investigated by comparing the test results of Runs 922 and 932. The ADS flow area in Run 932 was reduced to a half of that in Run 922 which was the reference test with a 5% break.

The system pressures in the two tests are compared in Fig. 7.1. The ADS was actuated at nearly the same time in the two tests. The depressurization rate after the ADS actuation was smaller in Run 932 than in Run 922 and the actuations of LPCS and LPCI in Run 932 were delayed by 96 and 166 s, respectively, compared to Run 922 as shown in Table 7.2.

The liquid levels in the pressure vessel in two tests are compared in Fig. 7.1. When the ADS was actuated, the downcomer was almost empty and a top quarter of the core was uncovered in the two tests.

After the ADS actuation the LPF initiated and the mixture level in the core swelled over the upper tie plate (UTP) in Run 922, whereas in Run 932 the LPF was weaker and the level swelled just to the UTP. The whole core was uncovered earlier in Run 922 than in Run 932 after the LPF subsided. The discontinuity of the liquid level occurred at the lower tie plate (LTP) in Run 922 because of CCFL, which was unclear in Run 932. The difference between the two tests was caused by the difference in steam generation rate inside core shroud due to flashing and the resultant difference in steam upflow velocity at LTP because the ADS flow area in Run 932 was one half of that in Run 922.

The mixture level in the core recovered to the UTP a little after the LPCI actuation in both tests. The durations of the core uncovering at the middle plane of the core were 211 and 355s in Runs 922 and 932, respectively. The liquid level in the downcomer recovered within 20 s after the recovery of the mixture level in the core.

The PCTs in the two tests are compared in Fig. 7.1. The PCTs were observed at the middle plane of the core and were 830 and 951 K in Runs 922 and 932, respectively. The PCT in Run 932 was recorded shortly after the LPCS actuation, while after the LPCI actuation in Run 922. The low steam discharge rate from the ADS in Run 932 restricted the steam upflow in the core and resulted in the increase of the falling rate of the LPCS water in the core. The heat transfer coefficients after the LPCS actuation were about 70 W/m²K for Run 922 and about 85 W/m²K for Run 932 at the middle plane of the fuel rod which recorded the PCT.

The primary effect of the ADS area reduction in the ROSA-III test was the actuation delay of the LPCS and LPCI, which resulted in a higher PCT.

7.3.2 Effect of ADS Actuation Timing

The effect of early actuation of ADS was investigated by comparing the test results of Runs 921 and 931. Both tests had a 1% break area at the recirculation pump suction line. The trip level in the downcomer to actuate the ADS was raised from the normal L1 level in the reference test Run 921 to L2 in Run 931 to actuate the ADS earlier.

The ADS was actuated earlier by 92 s in Run 931 than in Run 921 as shown in Fig. 7.2. The depressurization rate after the ADS actuation was larger in Run 921 than in Run 931 because of the less mass inventory and smaller core power in Run 921 at the ADS actuation time than in Run 931. Therefore, the differences in the LPCS and LPCI actuation times between the two tests were shortened to 44 and 31 s, respectively, compared with 82 s for the ADS.

The liquid levels in the pressure vessel in the two tests are compared in Fig. 7.2. The
liquid level in the downcomer at the ADS actuation timing was above the UTP level inside shroud in both tests because of a smaller break area of 1% than 5% in Runs 922 and 932. After the ADS actuation the mixture level in the downcomer swelled temporarily and decreased thereafter. However, the mixture level in the downcomer did not fall below the UTP level throughout the tests.

The mixture level inside the shroud in both tests was in the upper plenum when the ADS was actuated. As the LPF subsided the mixture level in the core lowered and the whole core was uncovered with the temporary acceleration of level fall after the FWF. The mixture level in the core recovered only by the LPCS water before the LPCI actuation. The core mixture level transients in Fig. 7.2 are overlapped each other when the time axis was adjusted. Thus, there is littel difference in duration of the core dryout between the two tests.

The PCTs in the two tests are compared in Fig. 7.2. The PCT was observed at Position 3, 352.5 mm above the middle plane of the core and was 751 and 766 K in Runs 921 and 931, respectively. The PCTs in both tests were recorded shortly after the LPCS actuation. The temperature increasing rates and durations in the two tests during the heatup period were nearly the same. Therefore, the PCTs in the two tests were similar values. This small difference between the PCTs of the two tests was the result of the small difference of the histories of the mixture level transients in the core between the two tests.

The earlier actuation of the ADS by L2 signal with a delay of 120 s had littel influence on the core cooling during a SBLOCA.

7.4 Analysis

The capability of the THYDE-B1/MOD2 code5) to calculate the fundamental thermal-hydraulic behavior of the system pressure, the liquid levels inside and outside the shroud, and the cladding surface temperatures was examined by analyzing the ROSA-III SBLOCA tests. The calculations were performed with the same models and node distribution in the ROSA-III and BWR analyses to study the similarity of the fundamental phenomena between the ROSA-III and the BWR during a SBLOCA by comparing the calculated results. The BWR SBLOCA was analyzed to investigate the ADS effect on the core cooling in a SBLOCA and to study better conditions for the ADS actuation.

7.4.1 Analytical Conditions

The node and junction representation of the ROSA-III is shown in Fig. 7.3. The primary models used to analyze ROSA-III tests and BWR SBLOCAs are compared in Table 7.3. Two three-region node volumes were used to represent the inside and outside regions of the core shroud in the pressure vessel. The pipes for the two recirculation loops were represented by four homogenous volumes. The jet pumps (JPs) were expressed by three junctions. The volumes of the JPs were added to those of the downcomer. The mixture levels in the two three-region node volumes were calculated by the Wilson's bubble rise model6) and the dynamic bubble sweepout length model of the THYDE-B1/MOD2. The mixture level in the downcomer was used as the trip levels L3, L2 and L1 to control the MSIV closure and the ECCS actuations. The initial mixture level in the downcomer was L3 in both ROSA-III and BWR calculations. The heater rods in the core were represented by three groups of heat slabs, one of which corresponded to the peak power rods in the peak power channel to calculate the PCT. The Moody's model7) and orifice model were used for the critical flow model at the break and the ADS. The former model was used for the saturated mixture or vapor with the discharge coefficient ($C_D$) of 0.61 and the latter for the subcooled liquid region with the $C_D$ of
0.60. The LPCS and the LPCI were injected into the upper plenum and the lower plenum, respectively, with a proper mixing length and a mixing coefficient for the steam and mixture regions.

After the dryout of the heater rods the steam and spray heat transfer models were used. The heat transfer coefficient (HTC) of 40 W/m²K was used as the steam cooling HTC during core heatup and 70 W/m²K as the spray HTC after the LPCS actuation. These HTC values were obtained from the ROSA-III SBLOCA tests\(^5\). When the heater rods were covered with the mixture the built-in heat transfer correlations were used.

### 7.4.2 Analyses of ROSA-III Tests

Four ROSA-III tests (Runs 922, 932, 921 and 931) were analysed with the THYDE-B1/MOD2 code. The experimental data were used in the analysis for the streamline flow rate, the LPCS and LPCI flow rates as a function of the pressure, and the feedwater flow rate as a function of time. The initial pressure, enthalpy and void fraction of each volume and the flow rate at each junction were given to be consistent with the experimental conditions before the break. The capability of the THYDE-B1 code to predict the BWR LOCA was discussed mainly for two 5% break tests.

The trends of the pressure histories after the break were basically the same among the four tests as mentioned in Section 7.3. The pressure decreased after the break, increased after the MSIV closure, stayed constant by the SRV, and decreased rapidly after the ADS actuation.

The calculated system pressures are compared with the test results in Fig. 7.4 for the 5% break tests. The calculated vessel pressures agreed well with the measured ones. The temporary pressure hold after the initiation of FWF was not calculated because the feedwater line was not modeled in the analyses. The calculated pressure decreased faster after the LPCS actuation because of the steam condensation in the upper plenum and increased after the reflooding because of the steam generation in the core. Due to faster pressure decrease after the LPCS actuation in the THYDE-B1 calculation, the LPCI was actuated earlier in the calculation than in the test.

The calculated times of MSIV closure and ADS actuation coincided well with the data showing that the liquid level drop in the downcomer agreed well with the data.

The calculated mixture level inside core shroud is compared with the data in Fig. 7.4. The overall agreement was good between the calculated results and the data. However, the mixture level drop in the core before the ADS actuation was larger and the mixture level swell after the LPF was smaller in the THYDE-B1 calculation than in the test for a 5% break. The differences between the calculated results and the measured data in the core reflooding behavior were due to the differences in actuation times and flow rates of the ECCS caused by the differences in pressure behaviors between calculated and measured results.

The fuel rod surface temperature transients are shown in Fig. 7.4 for the positions where the PCT is recorded. The calculated temperature behavior itself agreed well with the data because the calculated pressure and mixture level in the core agreed well with the data. The calculated PCTs were a little higher than the measured ones. The PCT was calculated higher by the reduction of the ADS flow area in accordance with the trend of the data.

The PCTs were recorded at Position 4 except for the calculated PCT in a 5% break with the 100% ADS flow area, which occurred at Position 3 and at the time of the LPCS actuation. The maximum temperature at Position 4, however, was calculated after the LPCI actuation as in the experiment.

The good agreement between the calculated temperatures and the data indicates the
appropriateness of the two HTCs used in the analyses for the steam and spray cooling.

In a 1% break the calculated system pressure, mixture level in the core and the fuel rod surface temperature agreed well with the data. There was little difference between the calculated PCTs for the two tests with different ADS actuation times as in the experiment.

The THYDE-B1/MOD2 code was able to calculate well the trends of the primary thermal-hydraulic behaviors of the system pressure, the mixture level in the core and the heater rod surface temperature observed in the four ROSA-III SBLOCA tests.

7.4.3 Similarity Analysis between ROSA-III Test and BWR LOCA

The similarity between the ROSA-III test and a BWR LOCA was investigated in the fundamental thermal-hydraulic phenomena through the THYDE-B1/MOD2 analyses. The same models were used in the BWR SBLOCA calculations as used in the ROSA-III analyses. However, the some calculation conditions are different in the area where the characteristics are different between the ROSA-III and BWR analyses as shown in Table 7.3. The calculated results of the ROSA-III tests and BWR LOCAs are compared in Figs. 7.5 and 7.6 for the 5% and 1% break LOCAs, respectively.

The overall trend of the pressure after the break was the same between ROSA-III and BWR in both 1% and 5% break LOCAs. The following characteristics were observed in the ROSA-III calculation results compared to the BWR SBLOCA results.

1) The pressure decreased a little slower after the break mainly because of conservative core power curve in a ROSA-III test.
2) The pressure increased more faster after the MSIV closure because the calculated linear heat transfer rate in the core was conservative in the ROSA-III and it was about two times that of a BWR at 25 s after the break.
3) The depressurization rate after the ADS actuation was smaller especially in the 1% break test because the amount of the stored heat release to the liquid in the ROSA-III pressure vessel was more than three times that in BWR when the stored heat is normalized by the volumetric scaling ratio (1/424). The small difference between the ROSA-III and BWR results in a 5% break was due to the empty downcomer and little heat transfer from wall to steam at the ADS actuation time.

The ADS was actuated fairly later in the BWR 1% break LOCA than in ROSA-III as shown in Fig. 7.6. The liquid level in the downcomer decreased more slowly from L2 to L1 level in BWR than in ROSA-III because the liquid levels inside and outside core shroud were well balanced in the BWR 1% break LOCA during the liquid level fall from L2 to L1, but were not balanced in ROSA-III and other BWR LOCAs. The times at which the liquid level lowered to L2 and L1 levels are compared in Table 7.5 for ROSA-III tests and BWR LOCAs.

The calculated core mixture level of the BWR was normalized by the length between the UTP and the LTP. The agreement was good for the mixture level behavior in the core in the 5% break between the ROSA-III test and BWR SBLOCA. However, the mixture level in the BWR was calculated higher than that of ROSA-III in the 1% break LOCA. There are two reasons for the higher mixture level calculated in the BWR 1% break LOCA. One is the faster depressurization rate of the BWR 1% LOCA than the ROSA-III and the other is the new mixture level calculation model of the THYDE-B1/MOD2 code in which the bubble in the mixture is assumed to be generated at a half height of the mixture region and rises with the Wilson's bubble rise velocity. The bubble sweep-out length to the mixture surface was longer in the BWR calculation than in the ROSA-III resulting in smaller bubble separation rate and the higher void fraction of mixture.

The temperature behavior from dryout to quench were similar between the BWR results
and the corresponded ROSA-III results in a 5% break in spite of the differences in the fuel rod structure and materials. The rod surface temperature is closely related to the mixture level behavior in the core and the mixture level in the BWR core was calculated higher than that in the ROSA-III in a 1% break. The core was uncovered only at upper one third and the dryout period was short in a BWR 1% break LOCA especially with the early ADS actuation condition (L2 + 120 s), therefore, the PCT was very low compared with that in the ROSA-III.

The similarity between the BWR SBLOCA and the ROSA-III SBLOCA test was confirmed in the overall trend of the thermalhydraulic behavior after the break such as the system pressure, the mixture level in the core and the rod surface temperature during the dryout.

7.4.4 BWR SBLOCA Analyses with Different ADS Actuation Conditions

The calculation parameters were the ADS actuation time, the ADS flow area and the break area. The effect of these parameters was discussed and the effectiveness of the current ADS actuation condition of the BWR was investigated. The study was mainly focused on the parameter effects on the PCT.

(1) The effect of the ADS actuation timing

The ADS actuation time was changed from the current L1 signal with a time delay of 120 s (L1 + 120 s) to an earlier actuation condition of L2 + 120 s, and to delayed ADS actuation conditions of L1 + 180 s, L1 + 600 s and no ADS. The break area was changed from 1% to 75% of the MRP suction pipe cross section.

The PCTs obtained from the BWR LOCA calculations are shown in Fig. 7.7. The PCT spectrum obtained from the ROSA-III tests in the break area parameter test series with the HPCS single failure\(^3\)\(^9\) is also presented in the figure for comparison.

In early actuation cases of ADS, the PCT decreased little, within 31 K from the standard case except for the 1% break LOCA. The PCT in the 1% break LOCA was lowered considerably (143 K) by the early ADS actuation and the early actuation of LPCS. There is a large time difference between L2 and L1 signals in a 1% break. The PCT decrease in a 1% break by the early ADS actuation was also observed in the ROSA-III calculation as mentioned in Section 7.4.2.

In the cases of time delay less than 60 s in ADS actuation, L1 + 180 s, the PCT changed little especially in the cases of break areas below 15%.

The PCT increased considerably by a large delay in ADS actuation, L1 + 600 s. The PCTs in the case of no ADS actuation were calculated which were higher than those in the case of L1 + 600 s for break area below 15%, but the difference between the two cases was small. However, the actual PCT would be higher in these cases. The saturation temperature is used as a steam temperature in the THYE-B1 code, but the heat flux at the rod surface should be overestimated when the steam is estimated to be in the superheated condition.

The PCT in the case of no ADS actuation decreased with an increase in the break area and approached to that calculated with the current ADS actuation condition when the break area became 50% of the recirculation pump suction pipe cross section.

The current ADS actuation timing (L1 + 120 s) of a BWR was assured to be effective for the core cooling under failed HPCS conditions. A little change in the ADS actuation time had little effect on the core cooling.

(2) The effect of the ADS flow area

The SBLOCAs with break areas of 1%, 5%, 15% and 25% of the MRP suction pipe cross section were analyzed changing the ADS flow area from 50% to 150% of the standard case.

The obtained PCTs are shown in Fig. 7.8. The PCT spectrum obtained from the ROSA-III tests is also presented in the figure for comparison.
With decrease in ADS flow area, the calculated PCT increased. This tendency was also observed in the ROSA-III tests and calculations as shown in the figure. The ADS actuation has two effects on core cooling, as mentioned in Introduction. On one hand the ADS improves the core cooling by actuating the LPCS and the LPCI by depressurization and on the other hand the ADS deteriorates the core cooling by mass discharge from the pressure vessel through ADS line. It has been confirmed from the ROSA-III test results and BWR LOCA analyses that the faster depressurization after the ADS actuation with the increased ADS flow area is more effective in the core cooling even if a larger mass is discharged from the ADS before LPCS actuation.

7.5 Conclusions

The ROSA-III experiments simulating 1% and 5% BWR SBLOCAs were performed with different ADS actuation conditions. These ROSA-III experiments and the corresponding SBLOCAs of the reference BWR were analyzed with the THYDE-B1/MOD2 code. The single failure of the HPCS was assumed in the experiment and the calculation. The following conclusions were obtained for a SBLOCA without HPCS actuation.

1) The primary thermalhydraulic behavior, such as the transients of the system pressure, the liquid level in the core and the heater rod surface temperatures, observed in the ROSA-III experiment were calculated well by the THYDE-B1 code.

2) The similarity was confirmed between the ROSA-III SBLOCA test and a BWR SBLOCA in the trends of primary thermalhydraulic behavior by using the THYDE-B1 code.

3) The effectiveness of ADS was confirmed for core cooling during a SBLOCA. The ADS is important in SBLOCAs with break areas below 50% of the cross-section of the recirculation pump suction pipe. For a break greater than 50%, the ADS becomes unimportant because the system is depressurized rapidly by a break flow before the actuation of ADS.

4) The ADS flow area has a significant influence on the core cooling. A decrease in the ADS flow area causes the delay of the LPCS and LPCI actuations and a resultant higher PCT. An increase in the ADS flow area results in a lower PCT on the contrary.

5) A little change in the actuation time of ADS has little influence on the core cooling and PCT. However, a large delay of the order of 10 minutes in the ADS actuation increases the PCT considerably.

References


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Fig. 7.6 Comparison of calculated major transient parameters in ROSA-III and BWR 1% break LOCA
Fig. 7.7 Peak cladding temperatures when ADS actuation time changes
Fig. 7.8 Peak cladding temperatures when ADS flow area changes
### Table 7.1 Test conditions of ROSA-III tests

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* at recirculation pump suction line  
** corresponds to 34% recirculation line break area

### Table 7.2 Sequence of events of ROSA-III tests

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<td>L1 signal</td>
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<td>ADS actuation</td>
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<td>Lower plenum flashing</td>
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### Table 7.3 Primary calculation conditions used in analyses

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<td>Number of junction</td>
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### Table 7.4 Time of L2 and L1 signals in ROSA-III and BWR

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* No calculation results
Fig. 7.1 Comparison of major transient parameters in ROSA-III 5% break LOCA test (Runs 922 and 932).
Fig. 7.2  Comparison of major transient parameters in ROSA-III 1% break LOCA tests (Runs 921 and 931).
Fig. 7.3 Node and junction representation of ROSA-III.
Fig. 7.4 Comparison of measured and calculated major transient parameters in ROSA-III 5% break LOCA tests.
Fig. 7.5 Comparison of calculated major transient parameters in ROSA-III 5% break LOCA.
Fig. 7.6 Comparison of calculated major transient parameters in ROSA-III 1% break LOCA.
Fig. 7.7 Peak cladding temperatures when ADS actuation time changes.

Fig. 7.8 Peak cladding temperatures when ADS flow area changes.
8. Effect of Pressure Control System

8.1 Introduction

A pressure control system failure test series was carried out at the ROSA-III facility to evaluate the effect of the pressure control system\(^1\) on thermal-hydraulic behavior during a BWR LOCA. In the BWR/6, the pressure control system maintains the turbine inlet pressure at 6.7 MPa by regulating the turbine throttle valve. If the turbine throttle valve is widely open and the pressure is still rising, the turbine bypass valve opens to control the pressure. In the ROSA-III tests, the pressure control system was simulated by a pressure control valve in the steam line (CV-130 ; see Fig. 2.2) and operated so as to maintain the vessel steam dome pressure above 6.7 MPa by regulating this valve.

This chapter presents results of LOCA tests with and without pressure control system. The severest single failure assumption of ECCS for core cooling is the failure of HPCS irrespective of the break area in the ROSA-III recirculation pump suction line break (see Appendices III and IV). The HPCS was also assumed to be inactive in the present test series. The break location was at the recirculation pump suction line in all the eight LOCA tests and the break areas were selected as follows.

It was already proved from the previous ROSA-III test results that the pressure control system did not operate in recirculation pump suction line breaks greater than 5% in break area due to actuation of the MSIV earlier than the pressure control system (see Chapter 5). Therefore, four break areas of 5, 2, 1 and 0% were selected. The primary objectives of the present test series are

1. to provide baseline information on thermal-hydraulic behavior during a BWR LOCA with pressure control system failure, and
2. to investigate the effect of the pressure control system on thermal-hydraulic phenomena during a BWR LOCA.

8.2 Test Conditions and Test Procedure

The major test conditions of the eight LOCA tests are listed in Table 8.1. The operability of the pressure control system is represented by "O (operation)" and "F (failure)" in the Run number. The zero percent break LOCA tests were carried out to investigate LOCAs with extremely small break areas. The simulated break was a split break with break orifices in the recirculation pump suction line.

The initial test conditions in all the tests were similar as follows. The steam dome pressure was 7.35 MPa corresponding to a saturation temperature of 562 K. The core power was 3.96 MW corresponding to the maximum linear heat rate (MLHR) of 16.7 kW/m. The core inlet flow rate was 16 kg/s and the core outlet quality was estimated to be 0.14. The subcooling of the lower plenum fluid was 11 K.

The test was initiated by opening the blowdown valve (QOBV "B" ; see Fig. 2.2) located immediately downstream of the break orifice. The core power was reduced in accordance with the predetermined curve as mentioned in Section 2.3.

The closure of the feedwater line valve was initiated at 2 s and completed at 3 s after break. The closing time of the feedwater line valve was based on the safety evaluation guide
for BWR/6. Downcomer liquid level signals were used to initiate MSIV closure and ECCS actuations in the test. The initial liquid level in the downcomer was 5.0 m. The liquid volume in the downcomer below 5.0 m in ROSA-III, including the volumes in the jet pump suction line pippings, is scaled to the liquid volume below the scram level (L3 level) in the BWR/6. The L2 and L1 liquid levels in the ROSA-III downcomer were 4.76 and 4.25 m, respectively. Main steam isolation valve closure was initiated by the L2 level signal with a time delay of 3 s. The LPCS, LPCI and ADS were initiated by the L1 level signal with time delays of 40, 40 and 120 s, respectively. These time delays were used in the safety evaluation of BWR/6\(^1\). The injections of the LPCS and LPCI were further specified to initiate at system pressures below 2.2 MPa and 1.6 MPa, respectively.

8.3 Test Results and Discussions

8.3.1 5% Break

In the 5% break test with active pressure control system, the pressure control system operated during only short time period and pressure control system actuation had almost no effect on the fundamental thermal-hydraulic phenomena during a LOCA. Hence, detailed descriptions of 5% break test results are omitted.

8.3.2 2% Break

The system pressure transients in the tests R20 and R2F are compared in Fig. 8.1(a) and the timing of major events in both tests is summarized in Table 8.2. The main steam and ADS flow rates in the tests are presented in Fig. 8.1(b). In the test R2F, the MSIV closure and ADS actuation were delayed due to delay of L2 and L1 signals, respectively, as will be pointed out later. Hence, the system pressure transients in the tests R20 and R2F differed considerably from each other. However, the system pressure transients in these two tests almost agreed after the system pressures decreased below 4.0 MPa. This is because the total discharge mass through the main steam and ADS lines, until the system pressure decreased to 4.0 MPa, was almost the same between the test R20 and R2F, as shown in Fig. 8.1(b). A small difference in the break flow had almost no effect on the difference in the system pressure transient.

The collapsed liquid levels in the downcomer, and the mixture levels in the peak power channel and the downcomer are presented in Fig. 8.1(c). In the test R2F, the system pressure decreased below 6.4 MPa before MSIV closure and the fluid in the downcomer flashed. Due to this flashing, the two-phase mixture level in the downcomer was kept high until the MSIV closed. Hence, the upper downcomer collapsed liquid level was also kept high before MSIV closure, while the sum of the upper and lower downcomer collapsed liquid levels was almost the same as that in the test R20. This led to the delay of L2 and L1 signals, which resulted in the delay of MSIV closure and ADS actuation, respectively. In the test R2F, after the MSIV closed, the upper downcomer collapsed liquid level decreased rapidly due to the collapse of bubbles in the downcomer caused by rapid pressure increase due to MSIV closure. The delay of MSIV closure in the test R2F resulted in mass discharge larger than in the test R20 (see Fig. 8.1(b)). Hence, the collapsed liquid level in the downcomer in the test R2F was kept somewhat lower than that in the test R20 after approximately 120 s.

In the test R2F, top of the peak power channel core was also uncovered temporarily due to bubble collapse caused by rapid pressure increase due to MSIV closure. Temporary uncovering at the top of the peak power channel core was observed again before the second lower plenum flashing in the test R2F. The difference in the mixture level transient of the
peak power channel core between the test R20 and R2F was small after mitigation of the second lower plenum flashing. The reason is that the mass inventories in the pressure vessel in the two tests were almost the same after mitigation of the lower plenum flashings as shown above. The following two factors are possible for the same mass inventories. First, the mass discharge from the system after mitigation of the second lower plenum flashing was almost the same between the two tests due to the offset of larger mass discharge through the main steam line and smaller mass discharge through the ADS line in the test R2F (see Fig. 8.1(b)). Second, the LPCS and LPC1 actuation times were almost the same between the two tests due to nearly the same system pressure transients below approximately 4.0 MPa.

The surface temperatures of the fuel rod A-11 in the tests R20 and R2F are compared in Fig. 8.1(d) and the PCTs in both the tests are summarized in Table 8.2. In the test R2F, temporary dryouts of the fuel rod surface, which were expected to occur by the temporary uncovering immediately after MSIV closure and before the second lower plenum flashing, was observed only at Position 1 of the fuel rod A-11. There were no significant differences in the fuel rod surface temperature transients and the PCT between the test R20 and R2F. This is due to small differences in the timing of major uncovering and reflooding, and the timing of LPCS and LPC1 actuations between the tests R20 and R2F. Some differences in quenching times between the tests R20 and R2F, shown in Fig. 8.1(d), are probably due to a difference in injected ECC water distribution between the two tests and slightly earlier actuations of LPCS and LPC1 in the test R2F. The fuel rod surfaces were quenched both downward from the core top and upward from the core bottom by LPCS and LPC1 water in both tests R2F and R20.

8.3.3 1% Break

The system pressure transients in the tests R10 and R1F are compared in Fig. 8.2(a) and the timing of major events in both the tests is summarized in Table 8.2. The main steam and ADS flow rates in the tests are presented in Fig. 8.2(b). As was the case with 2% break, in the test R1F, the MSIV closure and ADS actuation were delayed due to delay of L2 and L1 signals, respectively. Also as with 2% break, the system pressure transients in the tests R10 and R1F were similar after the system pressures decreased below approximately 4.0 MPa. However, the system pressure in the test R1F below approximately 4.0 MPa was kept somewhat lower than that in the test R10. The reason may be that, after the system pressure decreased below approximately 4.0 MPa, the total discharge mass until a certain time through the main steam and ADS lines in the test R1F was somewhat larger than that in the test R10, as shown in Fig. 8.2(b).

The collapsed liquid levels in the downcomer, and the mixture levels in the peak power channel and the downcomer are presented in Fig. 8.2(c). As was the case with 2% break, in the test R1F, due to the fluid flashing in the downcomer before MSIV closure, the collapsed liquid level in the downcomer (the upper downcomer collapsed liquid level) was kept high until the MSIV closed. This led to the delay of L2 and L1 signals, which caused the delay of MSIV closure and ADS actuation, respectively. Also as with 2% break, in the test R1F, the peak power channel core was uncovered temporarily due to bubble collapse caused by rapid pressure increase due to MSIV closure. The peak power channel core was uncovered somewhat earlier in the test R1F than in the test R10 after mitigation of the second lower plenum flashing. The reason may be that the mass discharge through the main steam and ADS lines before mitigation of the second lower plenum flashing was somewhat larger in the test R1F than in the test R10 (see Fig. 8.2(b)). The peak power channel core in the test R1F was also reflooded earlier than in the test R10 after ECCS actuations. This is because the LPCS
and LPCI were actuated earlier in the test R1F than in the test R10 due to faster depressurization in the test R1F.

The surface temperatures of the fuel rod A-11 in the tests R10 and R1F are compared in Fig. 8.2(d) and the PCTs in both the tests are summarized in Table 8.2. In the test R1F, the temporary dryout of the fuel rod surface arising from the temporary uncovering immediately after MSIV closure occurred at Positions 1 through 4 of the fuel rod A-11, whereas it was observed only at Position 1 in the test R2F (2% break test). The surfaces of the fuel rod A-11 dried out earlier in the test R1F than in the test R10 after mitigation of the second lower plenum flashing. The surfaces were also quenched earlier in the test R1F than in the test R10 after ECCS actuations. These earlier dryouts and quenchings of the fuel rod surfaces in the test R1F were consistent with the mixture level transient of the peak power channel core in the test. The fuel rod surfaces were quenched mainly downward from the core top only by LPCS water in both tests R1F and R10, while those were quenched downward from the core top and upward from the core bottom by LPCS and LPCI water in the 5 and 2% break tests. The PCT in the test R1F was 55 K lower than that in the test R10. This is due to shorter time period from core uncovering to LPCS actuation in the test R1F than in the test R10.

8.3.4 0% Break

The system pressure transients in the tests R00 and R0F are compared in Fig. 8.3(a) and the timing of major events in both tests is summarized in Table 8.2. The main steam and ADS flow rates in the tests are presented in Fig. 8.3(b). As were the cases with 2 and 1% breaks, the MSIV was closed considerably later in the test R0F than in the test R00 due to delay of L2 signal. However, contrary to this, the ADS was actuated earlier in the test R0F than in the test R00 due to earlier generation of L1 signal, as pointed out later. The system pressure in the test R0F was kept considerably lower than that in the test R00 throughout the transient. This is because the total discharge mass until a certain time through the main steam and ADS lines in the test R0F was considerably larger than that in the test R00 throughout the transient, as shown in Fig. 8.3(b).

The collapsed liquid levels in the downcomer, and the mixture levels in the peak power channel and the downcomer are presented in Fig. 8.3(c). As were the cases with 2 and 1% breaks, in the test R1F, due to the fluid flashing in the downcomer before MSIV closure, the collapsed liquid levels in the downcomer (the upper downcomer collapsed liquid level) was kept high until MSIV closure. This led to delay of L2 signal which resulted in delay of MSIV closure in the test R0F. However, since a larger amount of steam was discharged through the main steam line before MSIV closure in the test R0F (see Fig. 8.3(b)), the collapsed liquid level in the downcomer (the upper downcomer collapsed liquid level) in the test R0F became lower than that in the test R00 approximately 150 s after break. This led to earlier generation of L1 signal which caused earlier actuation of ADS in the test R0F. The peak power channel core was uncovered temporarily due to bubble collapse caused by rapid pressure recovery due to MSIV closure, as were the cases with 2 and 1% breaks. The peak power channel core was uncovered considerably earlier in the test R0F than in the test R00 after mitigation of the second lower plenum flashing. The reason may be that the mass discharge through the main steam and ADS lines before mitigation of the lower plenum flashing was considerably larger in the test R0F than in the test R00 (see Fig. 8.3(b)). The peak power channel core in the test R0F was also reflooded considerably earlier than in the R00 after ECCS actuations. This is because the LPCS and LPCI were actuated considerably earlier in the test R0F than in the test R00 due to faster depressurization in the test R0F.

The surface temperatures of the fuel rod A-11 in tests R00 and R0F are compared in
Fig. 8.3(d) and the PCTs in both tests are summarized in Table 8.2. In the test R0F, the temporary dryout of the fuel rod surface occurred immediately after MSIV closure at Positions 1 through 4 of the fuel rod A-11, as were the cases with the 1 and 2% break tests. The surface temperatures of several fuel rods rose due to this temporary dryout. The PCT in the test R0F was observed at Position 3 of the fuel rod A-77 during this temporary dryout. However, the PCT in the test R0F was lower by 42 K than that in the test R00. The PCT in the test R00 was observed at Position 3 of the fuel rod A-11 after mitigation of the lower plenum flashing as well as in the 5, 2 and 1% break tests. The surfaces of the fuel rod A-11 dried out considerably earlier in the test R0F than in the test R00 after mitigation of the second lower plenum flashing. The surfaces were also quenched considerably earlier in the test R0F than in the test R00 after ECCS actuations. These earlier dryouts and quenchings of the fuel rod surfaces in the test R0F were consistent with the mixture level transient in the test. The fuel rod surfaces were quenched downward from the core top only by LPCS water in both tests R0F and R00, as was the case with 1% break tests.

8.3.5 Effect of Pressure Control System

The timing of the key events is compared in Fig. 8.4 for all the eight tests. The PCTs in the tests are also presented in the figure. The maximum fuel rod surface temperature recorded after mitigation of the lower plenum flashing (594 K) is indicated in the figure as the PCT in the test R0F for easy comparison. (The real PCT in the test R0F (595 K) was observed during the temporary dryout before the lower plenum flashing as mentioned in Section 8.3.4.)

In a 5% break LOCA, the pressure control system operates during just short time period even if the pressure control system is active. Hence, pressure control system failure has almost no effect on the fundamental thermal-hydraulic phenomena during a LOCA.

In LOCAs smaller than 5% in break area, if the pressure control system is inactive, the upper downcomer liquid level is kept high until the MSIV closes due to fluid flashing in the downcomer before MSIV closure. This leads to the delay of L2 and L1 signals resulting in the delay of MSIV closure and ADS actuation, respectively except in a 0% break LOCA test R0F without pressure control system, in which the upper downcomer liquid level becomes lower than in a 0% break LOCA test R00 with pressure control system due to the discharge of a larger amount of steam through the main steam line before MSIV closure. This leads to earlier generation of L1 signal resulting in earlier ADS actuation in a 0% break LOCA without pressure control system.

The core is uncovered earlier with the inactive pressure control system after mitigation of the second lower plenum flashing in a LOCA with smaller break area than 2%. The reason may be that the mass discharge through the main steam and ADS lines is larger because of the inactive pressure control system. The core is also reflooded earlier in a LOCA without pressure control system. This is because the LPCS and LPC1 are actuated earlier due to faster depressurization in a LOCA without pressure control system. The PCT in a LOCA without pressure control system is lower than that in a corresponding LOCA with pressure control system. This is due to shorter time period from core uncovering to reflooding (or quenching) in a LOCA without pressure control system. The core is quenched only by LPCS water in 1 and 0% break LOCAs.

If the pressure control system is inactive, the top core is uncovered temporarily due to bubble collapse arising from MSIV closure. The fuel rod surface temperatures increase at the top part of core during this temporary uncovering especially in a 0% break LOCA. The timing of this temporary uncovering is not indicated in Fig. 8.4 to avoid complexity of illustration.
8.4 Conclusions

Four LOCA tests with pressure control system failure and corresponding four LOCA tests with intact pressure control system have been conducted at the ROSA-III facility to evaluate the effect of the pressure control system on fundamental thermal-hydraulic phenomena during a BWR small break LOCA with break areas between 0 and 5%. The HPCS was assumed to be inactive as the severest single failure of ECCS in all the tests. The break location was at the recirculation pump suction line. The following conclusions have been obtained.

(1) The pressure control system has no effect for a LOCA with a break area greater than 5%. In LOCAs with a break area less than 5%, the operation of the pressure control system prevents depressurization below the setpoint pressure of 6.7 MPa, delays the flashing and results in earlier MSIV closure. The fundamental thermal-hydraulic phenomena during a small break LOCA are significantly affected by the operation of the pressure control system.

(2) (a) If the pressure control system is inactive, the system pressure decreases below 6.4 MPa before MSIV closure and the downcomer fluid begins to flash. Mixture level swell in the downcomer due to this flashing delays considerably both the MSIV closure and ADS actuation, which are initiated by low upper downcomer liquid level signals (L2 and L1 signals, respectively).

(b) If the pressure control system is inactive, the core is uncovered temporarily after MSIV closure because of bubble collapse caused by pressure rise due to the MSIV closure and the fuel rod surface temperatures rise during this period especially in a LOCA with an extremely small break area.

(3) The PCT, which occurred mainly during the later core uncovering by boil-off, is lower with pressure control system failure than in a corresponding LOCA with intact pressure control system in a smaller LOCA than 2%. This is because the LPCS and LPC1 are actuated earlier in the LOCA with pressure control system failure due to earlier depressurization. In a LOCA with a break area larger than 2%, the PCT was not significantly affected by the operation of the pressure control system.

(4) The PCT is well below the present safety criteria of 1473 K even if both the pressure control system and the HPCS are assumed to be inactive.

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Fig. 8.2(a) Comparison of system pressure transients (1% break tests)
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Fig. 8.3(d) Comparison of fuel rod surface temperatures (0% break tests)
Fig. 8.4 Chronology of major events
### Table 8.1  List of tests

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<td>R5F (912)</td>
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* PCS : Pressure Control System

### Table 8.2  Comparison of major events and peak cladding temperatures

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<th>R1F</th>
<th>R00</th>
<th>R0F</th>
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<tr>
<td>PCS$^1$ Actuation (s)</td>
<td>21  25</td>
<td>25  32</td>
<td>23  34</td>
<td>22  35</td>
<td>21  25</td>
<td>25  32</td>
<td>23  34</td>
<td>22  35</td>
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<tr>
<td>MSIV Closure (s)</td>
<td>25  24</td>
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<td>34  101</td>
<td>35  126</td>
<td>25  24</td>
<td>32  87</td>
<td>34  101</td>
<td>35  126</td>
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<td>86  119</td>
<td>103 119</td>
<td>123 160</td>
<td>123 160</td>
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<tr>
<td>L2 Signal (s)</td>
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<td>28  85</td>
<td>30  92</td>
<td>30  123</td>
<td>21  19</td>
<td>28  85</td>
<td>30  92</td>
<td>30  123</td>
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<tr>
<td>L1 Signal (s)</td>
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<td>68  115</td>
<td>112 135</td>
<td>235 159</td>
<td>42  38</td>
<td>68  115</td>
<td>112 135</td>
<td>235 159</td>
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<td>Actuation of SRV$^2$ (s)</td>
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<td>84  99</td>
<td>71  188</td>
<td>79  231</td>
<td>85  355</td>
<td>75  122</td>
<td>84  99</td>
<td>71  188</td>
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<tr>
<td>ADS Actuation (s)</td>
<td>162 158</td>
<td>188 235</td>
<td>231 256</td>
<td>355 282</td>
<td>162 158</td>
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<td>204 184</td>
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<td>685 504</td>
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<td>363 347</td>
<td>471 460</td>
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<td>719 531</td>
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<td>531 514</td>
<td>618 568</td>
<td>773 588</td>
<td>426 406</td>
<td>531 514</td>
<td>618 568</td>
<td>773 588</td>
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**PCT**

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<td>Pos. 4</td>
<td>Pos. 3</td>
<td>Pos. 3</td>
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<td>540</td>
<td>519</td>
<td>547</td>
<td>496</td>
<td>696</td>
<td>153</td>
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</tbody>
</table>

1) PCS : Pressure Control System
2) SRV : Safety Relief Valve
3) LPF : Lower Plenum Flashing
4) JP : Jet Pump
5) MRP : Main Recirculation Pump
6) FWF : Feedwater Flashing
7) Reflooding of Core (Bottom ~ Top) : Reflooding of Core (Bottom ~ Top)

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* PCS : Pressure Control System

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No : No Recovery

* Occurring time in peak power channel
Fig. 8.1(a) Comparison of system pressure transients (2% break tests).

Fig. 8.1(b) Comparison of main steam and ADS flow rates (2% break tests).
Fig. 8.1(c) Comparison of liquid level transients (2% break tests).
Fig. 8.1(d) Comparison of fuel rod surface temperatures (2% break tests).
Fig. 8.2(a) Comparison of system pressure transients (1% break tests).

Fig. 8.2(b) Comparison of main steam and ADS flow rates (1% break tests).
Fig. 8.2(c) Comparison of liquid level transients (1% break tests).
Fig. 8.2(d) Comparison of fuel rod surface temperatures (1% break tests).
Fig. 8.3(a) Comparison of system pressure transients (0% break tests).

Fig. 8.3(b) Comparison of main steam and ADS flow rates (0% break tests).
Fig. 8.3(c) Comparison of liquid level transients (0% break tests).
Fig. 8.3(d) Comparison of fuel rod surface temperatures (0% break tests).
Fig. 8.4  Chronology of major events.
15. Main Steam Line Break Test Series

15.1 Introduction

The BWR LOCA studies have been concentrated on the main recirculation line break (MRLB) LOCAs rather than the other pipe-line break LOCAs including the main steam line break (MSLB) LOCA. The primary reason of selecting the MRLB LOCA as the principal research area among various BWR LOCAs is that it has been considered to be the severest for core cooling due to the fast fluid mass depletion from the pressure vessel. The MSLB LOCA\textsuperscript{13} has not been considered to be severe for core cooling because of a faster depressurization which results in earlier ECCS actuations, and a larger amount of remaining fluid mass as a result of high break elevation (see Fig. 15.1) in comparison with the MRLB LOCA. However, the larger amount of remaining fluid mass in the MSLB LOCA, combined with steam discharge flow, results in a higher downcomer water level than in the MRLB LOCA. As the low downcomer water level trips to actuate important systems such as MSIV and ECCSs in the BWR system, the higher downcomer water level may disturb the operator's recognition of abnormal events as in the TMI-2 accident. The similar event may occur also in a transient by the abnormal turbine bypass valve opening.

The principal objective of this chapter is to clarify experimentally those characteristic features of the MSLB LOCA in comparison with those of the MRLB LOCA. A failure was assumed on the trip signal by high reactor containment vessel (RCV) pressure to investigate effects of the delayed downcomer level trip on the MSLB LOCA. A large MSLB LOCA experiment was performed at the FIST facility\textsuperscript{2} by assuming the failure of two LPCIs with respect to the experimental study of a BWR MSLB LOCA, six MSLB LOCA experiments\textsuperscript{3}–\textsuperscript{6} in the ROSA-III program, and a large MSLB LOCA test with an HPCS failure assumption at the TBL\textsuperscript{7} test facility. The 100\% MSLB tests with HPCS actuation at ROSA-III and FIST are investigated experimentally\textsuperscript{8} and analytically\textsuperscript{9}. The similarity of the ROSA-III test to a MSLB LOCA of a BWR/6 is evaluated in Chapter 20. Presented in this chapter, are the fundamental features of the MSLB LOCA compared to the MRLB LOCA, the effects of break area, break locations, and trip times of ECCS and MSIV.

15.2 Test Conditions and Test Procedure

Six MSLB LOCA tests have been performed at the ROSA-III facility (Fig. 15.2) as shown in Table 15.1. The break location is chosen at one of the four MSLs inside the RCV of the reference BWR for five tests and at the downstream-side of the MSIV outside the RCV for one test (Run 955). The break area is simulated by an orifice (OR-5) shown in Fig. 15.3 and is varied from 10\% to 100\% of the 1/424 scaled BWR MSL flow area. The main steam line is simulated by three branches as shown in Fig. 15.4. The third branch with a pressure control valve (CV-130) is used as a steady state steam line in a time period before MSIV closure and as a relief valve operation after the MSIV closure. The MSIV is simulated by the pressure control valve. The first branch with a break orifice and a quick opening air valve is used for a broken steam line. The second branch is used for an ADS flow.

The ECCSs are tripped by the following logics in the ROSA-III MSLB tests. A failure of the containment pressure signal for ECCS actuation is assumed in all the tests. Therefore,
the HPCS is actuated by a low downcomer water level trip (L2) with a time delay of 27 seconds, and the LPCS and 3 LPCIs are actuated by the "and"-signals of low downcomer water level trip (L1) with time delay of 40 seconds and the system pressure decreased below each injection pressure. The HPCS failure is assumed in all the tests except for Run 952. These mean that single failure of the RCV pressure trip is assumed in Run 952, and double failures of the RCV pressure trip and the HPCS are assumed in the other five tests. The ADS is actuated by a low downcomer water level trip (L2) with time delay of 120 s only in small MSLB tests (Runs 954 and 956) and in Run 955.

The MSIV is tripped to close at the break initiation in four tests, by L2 level trip with time delay of 3 s in Run 956 (similar as in BWR/5 system) and at 3 s after the break in Run 955. The effects of MSIV trip timing on the small MSLB transient phenomena are investigated by comparing the results of Runs 954 and 956.

The initial test conditions of the six MSLB tests are similar to those of the BWR/6 rated conditions. The lower initial core flow rate is established in the tests to preserve the same void distribution across the core as the reference BWR rated condition. The core power curve\textsuperscript{10} simulates a conservative heat transfer rate of BWR core in a calculated large MRLB LOCA except for the first nine seconds after the break.

The six MSLB LOCA tests are performed as follows. After establishing the steady initial test conditions, the data recording system is actuated. The break initiation is simulated by quickly opening the air valve (AV-165) in the first MSL branch and quickly closing the CV-130 as an MSIV closure. In Run 955 the pressure control valve (CV-130) is operated again as a relief valve after the MSIV closure. The setpoint pressure of the BWR relief valve is 8.13 MPa. At the break time, the main recirculation pumps (MRPs) initiate to coast down. The feedwater supply is terminated during 2 and 4 seconds after the break. The tests are terminated after completion of core refloodling and filling up the pressure vessel by water.

15.3 Test Results

Fundamental features of the MSLB LOCA are clarified by comparing the test results with those of the MRLB LOCA in 15.3.1. The reasons of the high downcomer water level are shown in 15.3.2. The effects of break area and ECCS on the MSLB LOCA are shown in 15.3.3 and 15.3.4, respectively.

15.3.1 Fundamental Phenomena of MSLB LOCA Compared to MRLB LOCA

(1) Large Steam Line Break LOCA

The results of a 100% MSLB LOCA test (Run 953) are compared with those of a 200% MRLB LOCA test (Run 926\textsuperscript{11}). These tests have the same initial test conditions and HPCS failure assumption. Moreover, the total effective choking flow area (see Chapter 14) of both tests are similar, i.e., 140\% of the scaled MRL flow area for the former and 121\% of that for the latter (sum of flow areas at the jet pump drive nozzles and the PV-side break).

Figures 15.5(a) through (c) and Table 15.2 compare the results of both tests. The recirculation line uncoverey (RLU) in Run 926 was followed by the lower plenum flashing at 17 s after the break. On the other hand, the LPF in Run 953 initiated earlier at 4.2 s after the break with faster depressurization than Run 926. The faster depressurization and the earlier LPF initiation are the characteristic features in the early blowdown phase of the MSLB LOCA compared with the MRLB LOCA.

Another important characteristic feature of the MSLB LOCA is a depressurization rate after the LPF initiation. As the steam discharge flow rate and the core power generation
rate during the following time period were similar in both tests, the depressurization was significantly affected by the remained water mass as follows. Namely, an average depressurization rate during the LPF initiation time (6.4 MPa) and the feedwater flashing initiation time (FWF, 2.2 MPa) in Run 953 was 44.4 kPa/s, which was 1/1.82 of that in Run 926. And the remained water mass at the LPF initiation time in Run 953 was 1.86 times larger than that in Run 926. The lowered depressurization rate resulted in higher system pressure in Run 953 in the later blowdown phase and reflooding phase in comparison with Run 926.

Figures 15.5(b) and (c) show the fuel rod surface temperatures which recorded the PCT, and the water levels in PV of Runs 953 and 926. The fuel rod dryout at the top of core (see Table 15.2), which was detected by the surface temperature increase, was earlier in Run 953 than in Run 926. The core mixture level detected by the conduction tubes at the inner surface of the channel box decreased more slowly in Run 953 than in Run 926. The delayed LPCI actuation which was caused by delayed L1 level trip in Run 953, resulted in 220 K higher PCT than in Run 926. The effect of LPCI was larger than that of LPCS in both tests.

The most characteristic features of a large MSLB LOCA in comparison with the MRLB LOCA are that the upper downcomer water level is maintained at high level for a long time and that the core mixture level is significantly lower than the upper downcomer water level. The reasons of the core mixture level lower than the downcomer water level in the large MSLB LOCA are the larger fluid mass in the downcomer, higher void fraction in the jet pumps and downcomer and larger friction loss at the separator as shown in the next section. On the other hand, the core mixture level in a large MRLB LOCA is kept higher than the downcomer water level due to contribution of water head in the jet pumps. It is also shown through the comparison of test results that the bottom core recovery is significantly delayed in the MSLB LOCA than in MRLB LOCA. This is also one of the characteristic features of the MSLB LOCA.

(2) Small Steam Line Break LOCA

Two 10% MSLB tests (Runs 954 and 956) are compared with the 15% MRLB test (Run 927) as shown in Figs. 15.6(a) through (c) and Table 15.3. The 10% MSL break area corresponds to the 14% MRI break area. The small MSLB LOCA is mainly controlled by the MSIV closure, the ADS actuation and the ECCSs actuations as in a small MRLB LOCA.

The similar pressure response as in Run 927 was observed in Run 954 with the pressure increase due to the MSIV closure at the break time, the rapid pressure decrease due to safety valve opening between 5.4 and 17 s after the break and the depressurization by the ADS actuation. On the other hand, the pressure in Run 956 decreased after the break until the MSIV closure, which lowered the depressurization rate. In spite of the similar choking flow areas and the same core power between Run 954 and Run 927, the depressurization rate of the former at 150 s after the break was smaller than that of the latter due to the difference in the remained fluid mass as shown in the large MSLB LOCA characteristics. The remained fluid mass was more than 2 times larger in the MSLB tests than in the MRLB test. The delayed MSIV closure at 129 s in Run 956 lowered the pressure and slightly decreased the remained fluid mass in comparison with Run 954. However, the remained fluid mass in both test was similar at the time after ADS actuation because the ADS in Run 956 actuated later than in Run 954.

In Run 927, a temporary fuel rod dryout occurred at the upper part of core at 87 s after the break by the core mixture level fall and the rods were rewetted by the swelled mixture level caused by the LPF initiated at 117 s. The whole core was uncovered by the mixture level fall at 226 s and quenched finally at 393 s after the break within 37 s after the LPCI
actuation. During the reflooding process, the PCT was observed 11 s after the LPCI actuation in Run 927.

On the other hand, the whole core was not uncovered in the two MSLB tests. The core quench was completed within 24 s from the LPCS actuation in Run 954 and 18 s from the LPCS actuation in Run 956. The LPCI had little influence on the PCT in the 10% MSLB tests. In the small MSLB tests, the slower mixture level fall was also observed as in the large MSLB test.

(3) Large Steam Line Break Outside RCV

A 100% MSLB outside the reactor containment vessel (RCV) became equivalent to a small MSLB with steam discharge flow by a safety/relief valve (SRV) operation after the MSIV closure and the pressure response was very close to that of a 2% MRLB test (Run 920) as shown in Figs. 15.7(a) through (c) and Table 15.4. The key events controlling the pressure responses of both tests are the MSIV closure, the SRV operation (shown by SR in figure) and the ADS actuation. The ADS of Run 955 was actuated at 150 s after the break by the L2 level trip with a time delay of 120 s, whereas that of Run 920 was actuated at 188 s after the break by the L1 level trip with the same time delay. If the ADS was tripped by the L1 level in Run 955, the actuation would be delayed more than 200 s from the test results. The LPF was initiated after the ADS actuation in both tests. The LPCS and LPCI in both tests were actuated at their respective pressure setpoints.

The fuel rods were dried out due to the core mixture level fall after the ADS actuation in both tests. The core mixture level decreasing rate in Run 955 was fairly smaller than in Run 920 as shown in Fig. 15.7(c). This tendency is common for the MSLB LOCA tests compared with the MRLB LOCA tests with the similar break areas. In Run 920, PCT was recorded at 96 s after the LPCS actuation, whereas it was recorded at 12 s after the LPCS actuation in Run 955. Therefore, the LPCS actuation is considered sufficient for rewetting the dryout core in the 100% MSLB outside the RCV like as in the very small MRLB without HPCS.

15.3.2 Higher Water Level in Downcomer than Inside Shroud

Reasons for the higher water level in downcomer than inside shroud are shown below by studying the pressure balance in the system. Figure 15.8 shows the measuring locations of the differential pressures in the pressure vessel. The upper downcomer water level (DL 3.90–6.04 m) was measured by D2. If there were no measuring errors in the data, the sum of the differential pressures outside the core shroud should agree with that inside core-shroud, namely, \(-D1 + D2 = D3\). The differential pressures D4 and D5 are measured between the upper plenum (UP) and the steam dome (SD) and between the lower plenum (LP) and UP, respectively.

The differential pressures D1 through D5 (= D3 – D4) for Run 953 are shown in Table 15.5. The value (D3 + D1 – D2) shows the discrepancy of the differential pressures between bottom and top of PV in two routes. The small values indicate that the pressures in PV are well balanced.

In the transient phase in Run 953, fluid in the system began to flash at 4.2 s after the break and thereafter the generated steam rose up through the two paths toward the MSL, i.e., (1) LP-Core, Bypass-UP-Steam Separator (SS)-SD and (2) MRL and LP-DC-SD. Along the first path, there was a fairly large pressure loss across the SS, which is one of the reason for the higher upper downcomer water level in a MSLB LOCA test. The second reason is due to a very small differential pressure at the jet pumps \((-D1)\) in the second upflow path between 20 and 300 s, which indicate that the jet pump was almost filled by steam. The \(-D1\) and the fluid density at the jet pump outlet were larger in the smaller MSLB tests.
The third reason is the higher void fraction in the downcomer, which increased up to 56% at 90 s after the break and was maintained approximately at that value for a long time in Run 953. The void fraction was related to the steam discharge flow area as shown in the next subsection.

### 15.3.3 Effects of Steam Discharge Flow Area

A steam discharge flow from PV is limited in the MSLB LOCA of a BWR system by a break area and operations of the MSIV, the SRV (or SV) and/or the ADS. In this section, the effects of the total steam flow area including these valve areas on the MSLB LOCA phenomena are discussed.

**Table 15.6** shows the total steam flow area in each transient phase in the ROSA-III MSLB test series. The system pressure response was controlled by the total steam flow area as shown in **Fig. 15.9(a)**. In smaller MSLB tests, the MSIV closure and the ADS actuation on the pressure response had significant effects on the pressure response.

The upper downcomer water levels in the MSLB LOCA tests, however, showed a similar trend within a difference of 0.5 m as shown in **Fig. 15.9(b)**. The similar trend of the upper downcomer water level is one of the important characteristic features of the MSLB LOCA. The faster mass discharge in a larger MSLB LOCA was compensated by the larger pressure loss across the SS, higher void fractions in the jet pumps and lower downcomer in comparison with the smaller MSLB tests as shown previously.

The average void fraction in the lower downcomer was obtained from the corresponding differential pressure data (DL 0.94 – 3.90 m) as shown in **Fig. 15.9(c)**. The void fraction began to rise rapidly after the lower plenum flashing initiation, stayed at a constant value for a long time corresponding to the total steam flow area, and decreased by initiations of the LPCS and LPCI.

A clear dependence of the maximum lower downcomer void fraction ($\alpha_{\text{max}}$) on the total steam flow area (A) is shown for each test in **Fig. 15.10** and **Table 15.7**. The nine data points are consistently represented by an empirical equation of $\alpha_{\text{max}}=17 \times A^{0.475}$, where A is in m². This relation is applicable to transient phase after flashing except for some short periods after the break, MSIV trip and ADS actuation. In BWR systems, some geometrical factors and heat release from the structural metals which are different from those of the ROSA-III facility, may change this equation slightly.

### 15.3.4 Core Thermal-Hydraulic Response and Effects of ECCS

**Figure 15.11** shows the collapsed water levels in the core estimated from the differential pressures inside core shroud (D3 and D5 in **Fig. 15.8**), the core mixture level, and distributions of dryout and quench times of heater rods in Run 951. A difference between the two collapsed water levels, L(D3) and L(D5), was due to the pressure loss across the SS in the former shown in the previous section. It is shown from these comparisons that (1) the core mixture level was more than 1.0 m higher than the collapsed water level of L(D5), (2) the collapsed water level L(D3) was very close to the core mixture level by chance, and (3) the dryout and quench times were widely distributed above the collapsed level of L(D3). The similar phenomena were also observed in the average power bundles as well as in other MSLB tests. Therefore, it is important to measure the differential pressures inside the core-shroud to estimate the core cooling conditions in a MSLB LOCA.

**Figure 15.12** shows that the fuel rods can be quenched only by the HPCS actuated at 94 s after the break in Run 952 and that they can be quenched by the LPCS and 3 LPCI injected at 281 s after the break in Run 953. In Run 952, however, the surface temperatures
at the middle height (P4) of the five rods, A-31, A-68, A-71, A-82 and A-85 in the peripheral region, showed dryout again until the manual actuation of LPC1 at 1017 s after the break (Fig. 15.13). These peripheral rods have the same power generation rate as A-11 rod. The above phenomena of the second dryout indicates the imperfectness of the core cooling capability only by HPCS.

The LPCS and 3 LPCI actuations in Run 953 resulted in delayed but complete core cooling compared with Run 952 as shown in the fuel rod surface temperature transients in Fig. 15.12. The outer surface temperatures of fuel rods in Run 953 showed subcooled temperature after quench. The total water flow rate of LPCS and 3 LPCI in Run 953 was 6.4 times larger than that of HPCS in Run 952. It is concluded that the HPCS actuation is fairly effective to suppress the fuel rod temperature increase and limit the PCT, whereas actuations of LPCS and LPCI are necessary for the complete core cooling in a MSLB LOCA test.

15.3.5 Discussion on Similarity of ROSA-III Test to BWR/6 MSLB LOCA

The similarity of the 100% MSLB LOCAs between the ROSA-III, FIST and BWR systems is discussed in Chapter 20. The fundamental thermal-hydraulic phenomena in the MSLBs were confirmed to be similar among the three systems by assuming the same volumetrically scaled initial mass inventory, core power and ECCSs flow rates, and the same trip logics for the MSIV closure and ECCSs actuations. The effects of the distortions in scaling, i.e., the system geometry including the volume and height of the scaled components and the structural metal heat release, slightly affected the phenomena in the magnitude and time scale. These conclusions are the same as those for the large (200%) and small (5%) MRLB LOCAs between the ROSA-III and the BWR/6 systems. It is expected that the similarity is good also between the ROSA-III small MSLB LOCA test and the phenomena in a BWR/6 system as in a small MRLB LOCA because the differences of the small LOCA phenomena between MSLB and MRLB tests are small.

15.4 Conclusions

Six main steam line break (MSLB) tests were conducted in the ROSA-III program with test parameters of break location, break area and ECCS actuation conditions. The characteristic features of the MSLB LOCA were compared with those of MRLB LOCA and were analyzed with special interest in the relation of pressure balance and void distribution in the vessel. Following are the major conclusions obtained.

1. The MSLB LOCA in the BWR system is clearly distinguished from the MRLB LOCA by larger fluid mass remained in the pressure vessel, earlier flashing initiation, slower depressurization after flashing initiation, higher downcomer water level, and slower core mixture level decrease. These difference become smaller in a smaller break LOCA. A 100% MSLB outside the RCV becomes equivalent to a very small MSLB, in which steam is discharged only through safety/relief valves after the MSIV closure.

2. The downcomer water level is kept higher than the core mixture level in the MSLB LOCA, whereas the downcomer water level is always lower than the core mixture level in the MRLB LOCA. The reason for the level behavior in a MSLB LOCA is due to a pressure balance between downcomer and shroud-inside in PV with a large pressure loss of up-rising flow through the steam separator and large void fractions in both jet pumps and downcomer.

3. The total flow area for steam discharge through the break, MSIV and ADS controls
not only the discharge fluid mass and depressurization rate, but also the average void fraction in the downcomer and the water level difference between the upper downcomer and the core. The maximum of the average void fraction ($\alpha_{\text{max}}$) in the lower downcomer in the ROSA-III test is well correlated to the total steam flow area ($A$ : m$^2$) as

$$\alpha_{\text{max}} = 17 \times A^{0.475}.$$  

(4) The effectiveness of HPCS is clarified to limit the PCT. However, the effectiveness of the HPCS actuation alone is not sufficient in a 100% MSLB for the stable long-time core cooling but the actuations of LPCS and LPCI are necessary.

References

5) Suzuki, M., et al. : BWR main steam line break LOCA tests RUNs 951, 954 and 956 at ROSA-III —

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<tbody>
<tr>
<td>Item</td>
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<tr>
<td></td>
</tr>
<tr>
<td>Break Conditions</td>
</tr>
<tr>
<td>Break Location in RCV</td>
</tr>
<tr>
<td>Break Area Ratio ((*) (\dagger))</td>
</tr>
<tr>
<td>ECCS Conditions ((*) (\dagger))</td>
</tr>
<tr>
<td>Filled ECCS</td>
</tr>
<tr>
<td>Trip for HPSCS</td>
</tr>
<tr>
<td>Trip for LPCS/LPCI</td>
</tr>
<tr>
<td>Trip for ADS</td>
</tr>
<tr>
<td>MSIV Trip Time</td>
</tr>
<tr>
<td>Initial Conditions</td>
</tr>
<tr>
<td>Steam Dome Pressure</td>
</tr>
<tr>
<td>Core Power</td>
</tr>
<tr>
<td>Core Flow</td>
</tr>
<tr>
<td>Upper Plenum Quality</td>
</tr>
</tbody>
</table>

\((*)\) A 100% break area is 1/424 scaled BWR MSL flow area, which is 40% larger than a main recirculation line (MRL) flow area.

\((\dagger)\) L1 and L2 is 4.25 and 4.76 m above PV bottom, respectively.
### Table 15.2 Comparison of major events in 100% MSLB test (Run 953) and 200% MRLB test (Run 926)

<table>
<thead>
<tr>
<th>Events</th>
<th>Time after Break (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Run 953</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>0.0</td>
</tr>
<tr>
<td>L. P. Flashing</td>
<td>4.2</td>
</tr>
<tr>
<td>Core Top Uncovery</td>
<td>29.</td>
</tr>
<tr>
<td>Core Bot. Uncovery</td>
<td>200.</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>281.</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>281.</td>
</tr>
<tr>
<td>PCT Occurrence</td>
<td>290.</td>
</tr>
<tr>
<td>Final Core Quench</td>
<td>438.</td>
</tr>
</tbody>
</table>

### Table 15.3 Comparison of major events in 10% MSLB tests (Runs 954 and 956) and 15% MRLB test (Run 927)

<table>
<thead>
<tr>
<th>Events</th>
<th>Time after Break (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Run 954</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>0.0</td>
</tr>
<tr>
<td>SRV Operation</td>
<td>5.4 - 17.</td>
</tr>
<tr>
<td>L. P. Flashing</td>
<td>81.</td>
</tr>
<tr>
<td>ADS Actuation</td>
<td>153.</td>
</tr>
<tr>
<td>Core Top Uncovery</td>
<td>300.</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>443.</td>
</tr>
<tr>
<td>PCT Time</td>
<td>454.</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>455.</td>
</tr>
<tr>
<td>Core Final Quench</td>
<td>467.</td>
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### Table 15.4 Comparison of major events in 100% MSLB test outside RCV (Run 955) and 2% MRLB test (Run 920)

<table>
<thead>
<tr>
<th>Events</th>
<th>Time after Break (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Run 955</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>3.2</td>
</tr>
<tr>
<td>SRV Operation</td>
<td>8. - 150.</td>
</tr>
<tr>
<td>ADS Opening</td>
<td>150.</td>
</tr>
<tr>
<td>L. P. Flashing</td>
<td>178.</td>
</tr>
<tr>
<td>Core Top Uncovery</td>
<td>330.</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>516.</td>
</tr>
<tr>
<td>PCT Time</td>
<td>528.</td>
</tr>
<tr>
<td>Final Core Quench</td>
<td>546.</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>591.</td>
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</table>
Table 15.5  Pressure balance in large MSLB test (Run 953)

<table>
<thead>
<tr>
<th>Time (s)</th>
<th>$-D_1$ (kPa)</th>
<th>$D_2$ (kPa)</th>
<th>$D_4$ (kPa)</th>
<th>$D_5 = D_3 - D_4$ (kPa)</th>
<th>$D_3 + D_1 - D_2$ (kPa)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>95</td>
<td>-7</td>
<td>43</td>
<td>50</td>
<td>5</td>
</tr>
<tr>
<td>10</td>
<td>20</td>
<td>21</td>
<td>28</td>
<td>17</td>
<td>4</td>
</tr>
<tr>
<td>20</td>
<td>8</td>
<td>16</td>
<td>11</td>
<td>18</td>
<td>5</td>
</tr>
<tr>
<td>50</td>
<td>5</td>
<td>13</td>
<td>9</td>
<td>14</td>
<td>5</td>
</tr>
<tr>
<td>100</td>
<td>1</td>
<td>12</td>
<td>6</td>
<td>10</td>
<td>3</td>
</tr>
<tr>
<td>200</td>
<td>0</td>
<td>9</td>
<td>4</td>
<td>8</td>
<td>3</td>
</tr>
<tr>
<td>300</td>
<td>4</td>
<td>9</td>
<td>4</td>
<td>10</td>
<td>1</td>
</tr>
<tr>
<td>400</td>
<td>20</td>
<td>21</td>
<td>10</td>
<td>33</td>
<td>2</td>
</tr>
<tr>
<td>500</td>
<td>22</td>
<td>32</td>
<td>18</td>
<td>40</td>
<td>4</td>
</tr>
<tr>
<td>Height (m)</td>
<td>2.71</td>
<td>3.23</td>
<td>1.75</td>
<td>4.19</td>
<td>0.0</td>
</tr>
</tbody>
</table>

Measuring regions for $D_1$ through $D_5$ are shown in Fig. 15.11.

Table 15.6  Transient steam flow area in ROSA-III MSLB tests

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Break Area (%)</th>
<th>Steam Discharge Area$^{(1)}$ (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Before Break</td>
<td>After Break</td>
</tr>
<tr>
<td>954</td>
<td>10.0</td>
<td>29.3</td>
</tr>
<tr>
<td>956</td>
<td>10.0</td>
<td>29.3</td>
</tr>
<tr>
<td>951</td>
<td>33.7</td>
<td>29.3</td>
</tr>
<tr>
<td>953</td>
<td>100.0</td>
<td>29.3</td>
</tr>
<tr>
<td>955</td>
<td>100.0</td>
<td>29.3</td>
</tr>
</tbody>
</table>

$^{(1)}$ The break area is normalized by the 1/424 scaled MSL flow area of BWR/6. The 100% break area is $7.55 \times 10^{-4}$ m$^2$ in ROSA-III test.

Table 15.7  Maximum void fraction of lower downcomer fluid related to total steam flow area

<table>
<thead>
<tr>
<th>Run Number</th>
<th>Transient Phase</th>
<th>Total Area (%)</th>
<th>$\alpha_{\text{max}}$ (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>952 After Break</td>
<td>100.0</td>
<td>56</td>
</tr>
<tr>
<td>2</td>
<td>953 After Break</td>
<td>100.0</td>
<td>56</td>
</tr>
<tr>
<td>3</td>
<td>951 After Break</td>
<td>33.7</td>
<td>35</td>
</tr>
<tr>
<td>4</td>
<td>954 Before ADS</td>
<td>10.0</td>
<td>(17)</td>
</tr>
<tr>
<td>5</td>
<td>954 After ADS</td>
<td>35.0</td>
<td>34</td>
</tr>
<tr>
<td>6</td>
<td>956 Before MSIV</td>
<td>32.9</td>
<td>(33)</td>
</tr>
<tr>
<td>7</td>
<td>956 MSIV to ADS</td>
<td>10.0</td>
<td>17</td>
</tr>
<tr>
<td>8</td>
<td>956 After ADS</td>
<td>35.0</td>
<td>36</td>
</tr>
<tr>
<td>9</td>
<td>955 After ADS</td>
<td>25.0</td>
<td>29</td>
</tr>
</tbody>
</table>

( ) indicates estimated value from transient response of void fraction.
Fig. 15.1 Break location in BWR system.

Fig. 15.2 Overview of the ROSA-III system.
Fig. 15.3  Details of break orifice.

Fig. 15.4  Main steam line schematic.
Fig. 15.5  Representative features of 100% MSLB test (Run 952) compared with 200% MRLB test (Run 926), (a) system pressure, (b) fuel rod temperature, (c) water level.
Fig. 15.6 Representative features of 10% MSLB tests (Runs 954 and 956) compared with 15% MRLB test (Run 927), (a) system pressure, (b) fuel rod temperature, (c) water level.
Fig. 15.7  Representative features of 100% MSLB test outside RCV (Run 955) compared with 2% MRLB test (Run 920), (a) system pressure, (b) fuel rod temperature, (c) water level.
Fig. 15.8  Differential pressure measurements in pressure vessel.
Fig. 15.9 Effects of steam flow area on MSLB LOCA phenomena. 
(a) pressure response, (b) downcomer water level, (c) downcomer void fraction.
Maximum Void Fraction in Lower Downcomer

\[ \alpha_{\text{max}} = 17 \times A^{0.475} \]

(A in m²)

Break Area Ratio \( A^* \) (%)
\( A^* = A/A_{100}, \quad A_{100} = 7.55 \times 10^{-4} \text{ m}^2 \)

Data 1 - 9 are shown in Table 15.7.

**Fig. 15.10** Maximum void fraction of lower downcomer fluid related with total steam flow area.

**Fig. 15.11** Dryout and quench times related with core water level responses.
Fig. 15.12  Effects of ECCS on fuel rod temperature.

Fig. 15.13  Imperfect cooling of highest power rods in 100% MSLB test (Run 952) with HPCS actuation.
9. Natural Circulation Test Series

9.1 Introduction

The natural circulation core cooling is important during small break LOCAs and transients with loss of forced-circulation core cooling. This chapter presents results from a series of natural circulation experiments conducted in the ROSA-III facility under low-power and reduced inventory conditions. An analytical model is proposed to explain the experimentally observed natural circulation modes. The model can predict well the two-phase mixture level inside the core shroud for quasi-steady state natural circulation conditions, and thus can be used to study the BWR natural circulation behaviors.

9.2 Test Conditions

Five natural circulation experiments were conducted in the ROSA-III facility. The test boundary conditions are summarized in Table 9.1.

All the tests, with the exception of R3-NC-5, were conducted with the same procedure to study systematically the dependence of natural circulation behavior on core power, pressure and downcomer level. The facility was initially operated at a stable natural circulation condition for a nominal downcomer water level (L3). At this time, the saturated water was supplied from the feedwater line to the downcomer to compensate the steam flow. The system pressure was kept constant at 7.35 or 2.06 MPa, and the bundle power was controlled at 0.63 or 1.8 MW (7.0 or 20.0% of the volumetrically-scaled nominal power). Then, the tests were initiated by isolating the feedwater flow. As the system mass inventory was lost through the steam line, the downcomer water level decreased very slowly at a rate of less than 0.012 m/s. Data were collected during such a quasi-steady state boil-off.

Experiment R3-NC-5 addressed the effect of bundle power on the core uncoverory inception condition for a system pressure of 7.35 MPa. The downcomer water level was controlled constant, above the top of jet pumps, while the bundle power level was decreased stepwise, obtaining quasi-steady natural circulation at 20, 10, 5, 4, 3, 2 and 1% of the scaled full power. The two-phase mixture level inside the shroud was measured by using conductivity probes.

9.3 Test Results

The two natural circulation flow paths in ROSA-III are shown schematically in Fig. 9.1. A primary natural circulation loop is formed by the downcomer, core, separator and jet pumps. This circulation path is available only when the mixture level inside the shroud is high enough to allow liquid spill-over from the separator into the downcomer. In addition to this, an internal natural circulation loop is formed inside the shroud, by the core and core bypass regions, as long as the core is covered by mixture. Then, water in the upper plenum is drained into the bypass and flows into the bottom of the bundles through the bypass leakage holes.

Figures 9.2 and 9.3 show the primary circulation flow rates (measured at the jet pump discharge line) versus downcomer level for different bundle powers and system pressures. As the downcomer level decreased the primary circulation rate decreased. When the down-
comer level became lower than the core top elevation, the jet pump flow rate became almost constant and approximately equal to the main steam flow. This indicates the breakdown of the primary natural circulation, due to the uncovering of the separator. However, the core remained covered by two-phase mixture and the internal natural circulation continued. This internal natural circulation mode continued until the vessel mass inventory was depleted to the point where the downcomer static head failed to maintain the core covered by mixture. Figure 9.4 shows the behaviors of the downcomer water level and the in-shroud mixture level during Experiment R3-NC-1. In this experiment the core uncovering initiated slightly before the uncovering of the jet pump suction. Once the jet pump suction uncovered, the core uncovering progressed rapidly since there was no longer liquid replenishment from the downcomer. These changes of the natural circulation modes are illustrated in Fig. 9.5.

Figure 9.6 shows the mixture level inside the shroud as a function of the bundle power during Experiment R3-NC-5. As the bundle power was reduced, the bundle void fraction decreased, and the two-phase mixture level inside the shroud dropped. The core was uncovered for bundle power less than 5% of the scaled full power, at a system pressure of 7.35 MPa, for a normalized downcomer level, $H_c / H_L$ of 0.77.

9.4 Analysis

Three distinct modes of natural circulation have been identified: the primary circulation mode, the internal circulation mode, and the open-loop (core uncover) mode. The transition conditions between these modes are known in terms of the two-phase mixture level inside the shroud, which is dependent on bundle power, downcomer level and system pressure. In this section, we analyze the internal natural circulation mode, which is characterized by two-phase mixture level located in the upper plenum as shown in Fig. 9.5, to define the core uncover criteria. In the following analysis the natural circulation flow is assumed to be a one-dimensional quasi-steady flow.

9.4.1 Conservation Relations

First we consider conservation relations for the primary and internal circulation loops shown in Fig. 9.1.

Under a steady state condition, the boil-off of the in-shroud mass inventory is replenished by the liquid inflow from the downcomer. Then, assuming that main steam flow is saturated steam without liquid entrainment, we obtain, from the energy balance across the core, the main steam flow rate

$$W = \frac{Q_o}{h_f + \Delta h_{LP}}$$  \hspace{1cm} (1)

where $Q_o$ is the core power, $h_f$ is the latent heat of evaporation and $\Delta h_{LP}$ is the lower plenum subcooling.

The overall mass balance in terms of the core flow rate, $W_o$ is

$$W = X_c W_o$$  \hspace{1cm} (2)

where $X_c$ is core exit quality.

Thus the internal circulation flow in the bypass region is

$$W_B = W_o - W = (1 - X_c) W_o$$  \hspace{1cm} (3)

To write the momentum equation in a simple form, we use the average mixture densities in the core, $\langle \rho_m \rangle_c$, and in the upper plenum, $\langle \rho_m \rangle_P$, as follows:
\[ \langle \rho_m \rangle_c = \frac{1}{H_c} \left[ \rho_f L^+ \int_0^{H_c} \rho_m(z) \, dz \right] \]
\[ \langle \rho_m \rangle_r = \frac{1}{H-H_c} \int_{H_c}^H \rho_m(z) \, dz \]

where \( \rho = \rho_f (1-\alpha) + \rho_g \alpha \) with the void fraction, \( \alpha \). Other notations are defined in Fig. 9.1: \( H_c \) is the core height, \( L \) is the length of non-boiling region and \( H \) is the mixture level inside the shroud.

We also define an integral flow loss coefficient
\[ K_i = \frac{f_i k_i}{d_t} + k_i + 2 d X \left[ \frac{\rho_f}{\rho_g} - 1 \right] \]

where the first through third terms in the right hand side represent the frictional loss, area-change loss and spatial acceleration flow loss for an increase of quality \( d X \), respectively, for the \( i \)-th component. The acceleration loss term assumes a homogeneous flow. The frictional and area-change flow losses are calculated using Ueda’s expression\(^1\) for the Martinelli-Nelson two-phase flow multiplier
\[ \Phi_i = 1 + C_i X \left[ \frac{\rho_f}{\rho_g} C_i^2 - 1 \right] \]

where
\[ C_1 = 1.3, \ C_2 = 0.85 \] for \( X < 0.5 \),
\[ C_1 = 1, \ C_2 = 0.9 \] for \( X \geq 0.5 \).

The momentum equations for the two loops are then written simply,
\[ \langle \rho_f - \rho_m \rangle_c H_c = \sum \frac{W_i^2}{2 \rho_f g A_t^2} \text{ internal} \]
\[ \rho_g (H - H_0) + \rho_f H_0 = \langle \rho_m \rangle_c H_c + \langle \rho_m \rangle_r (H - H_c) \]
\[ + \left[ \sum \frac{W_i^2}{2 \rho_g g A_t^2} \right] \text{ primary} \]

where \( H_0 \) is the downcomer level. The density of subcooled liquid is assumed to be the same as that of saturated liquid.

The void fraction are calculated using an empirical correlation developed by Cunningham and Yeh\(^2\):
\[ \alpha = 0.925 \left( \frac{\rho_g}{\rho_f} \right)^{0.239} \left( \frac{j_e}{j_{terr}} \right)^a \left( \frac{j_e}{j_f+j_f} \right)^{0.6} \]

where
\[ a = 0.67 \] for \( j_e/j_{terr} < 1 \),
\[ a = 0.47 \] for \( j_e/j_{terr} \geq 1 \),
\[ j_{terr} = 1.53 (\rho_f/\rho_g)^{0.25} \]

\( j_e, j_f \) and \( \sigma \) are vapor and liquid superficial velocities and surface tension, respectively.

We also use the Meyer and Wilson correlation\(^3\):
\[ \alpha = C \left( \frac{\rho_g}{\rho_f - \rho_g} \right)^{0.12} \left( \frac{j_e}{\sqrt{gD_{terr}}} \right)^a \left( \frac{D_{terr}}{d} \right)^{0.1} \left( \frac{j_e}{j_f+j_f} \right)^{0.6} \]

where
\[ C = 0.564, \ a = 0.67 \] for \( 0 < (j_e/\sqrt{gD_{terr}}) < 1.5 \),
\[ C = 0.619, \ a = 0.47 \] for \( 1.5 \leq (j_e/\sqrt{gD_{terr}}) < 10 \),
\[ D_{terr} = \frac{\sigma}{\sqrt{g(\rho_f-\rho_g)}} \]
The Cunningham-Yeh correlation (10), based on rod bundle data, is applicable to core void fraction calculation. The Meyer-Wilson correlation (11) is, based on data from vessels, is used to calculate the void fraction in the upper plenum.

The axial distribution of quality in the core is calculated from the energy balance in the core boiling region ($z > L$),

$$X(z) = \frac{Q_o}{h_{fg} W_o} f_s (z / H_c) - \frac{\Delta h_{LP}}{h_{fg} + \Delta h_{LP}} f_l (z / H_c)$$  \hspace{1cm} (12)

or using Eqs. (1) and (2),

$$X(z) = X_e \left[ 1 - \frac{\Delta h_{LP}}{h_{fg}} \right] f_s (z / H_c) - \frac{\Delta h_{LP}}{h_{fg}} f_l (z / H_c)$$  \hspace{1cm} (13)

where $f_s (z / H_c) = 0.5 \cdot (1 - \cos \pi z / H_c)$, for a cosinusoidal core heat flux profile.

### 9.4.2 Solution of the Equations

We can now solve the equations (8) and (9) for the unknowns $W_s$ and $H$, for given system pressure, bundle power, $Q_o$, downcomer level, $H_o$ and lower plenum subcooling, $\Delta h_{LP}$.

Since the bypass holes provide most of the pressure losses along the internal circulation loop, Eq. (8) is rewritten approximately,

$$\Delta \rho (z_c) H_c = \frac{k_b W_s^2}{2 \rho_f g A_f^2}$$  \hspace{1cm} (14)

where,

$$\Delta \rho = \rho_f - \rho_g.$$

$k_b$ is the area-change loss coefficient of the bypass leak holes. The average void fraction in the core is obtained by integrating Eq. (10),

$$\langle \alpha \rangle_c = 0.925 \left( \frac{\rho_g}{\rho_f} \right)^{0.24} \left( \frac{W}{\rho_g A_c h_{cr}} \right)^{0.5} f_s$$  \hspace{1cm} (15)

where

$$f_s = \frac{1}{H_c} \int_0^{H_c} \left( \frac{X}{X_e} \right)^{0.86} \left( \frac{\rho_f - \Delta \rho X_e}{\rho_g + \Delta \rho X_e} \right)^{0.4} dz$$

and $A_c$ is the bundle flow area.

Thus for the two-phase mixture level inside the shroud $H$, we obtain the following expression from Eq. (9),

$$\frac{H}{H_c} = \frac{1}{1 - \langle \alpha \rangle} \left[ \frac{H_o}{H_c} + \langle \alpha \rangle \right] - \frac{1}{\Delta \rho g H_c} \left( \sum_i K_i W_i^2 / \rho_f A_i \right)_{primary}$$  \hspace{1cm} (17)

The average void fraction in the upper plenum is obtained by Eq. (11) putting $j_f = 0$

$$\langle \alpha \rangle = C \left( \frac{\rho_g}{\rho_f} \right)^{0.12} \left( \frac{W}{\rho_g A_p D_p} \right)^{0.4} \left( \frac{D_{cr}}{d} \right)^{0.1}$$

where $A_p$ and $d$ are the flow area and the hydraulic diameter of the upper plenum, respectively.

Eq. (17) is meaningful for $H > H_c$. For $H = H_c$, Eq. (9) reduces to a core uncover criterion in terms of downcomer liquid level

$$\frac{H_o}{H_c} = (1 - \langle \alpha \rangle) + \frac{1}{\Delta \rho g H_c} \sum_i K_i W_i^2 / \rho_f A_i^{primary}$$  \hspace{1cm} (18)

Obviously, the first term in the right hand side represents the gravitational head differ-
ence between the downcomer and core, and the second term represents the dynamic pressure loss.

### 9.5 Assessment of Model with Test Data

The present analytical model has been used to predict the ROSA-III experimental results. The primary and internal circulation flow rates predicted by the present model are shown in Figs. 9.7 and 9.8, together with the measured jet pump flow rates, for pressures of 7.35 and 2.06 MPa, respectively. The model assumes that the jet pump flow is equal to the boiloff rate of the in-shroud mass inventory. This occurs after the cessation of the primary circulation i.e., for downcomer water levels below the top of core, when the core power is less than 20%. The internal circulation flow through the bypass increases non-linearly with power.

Figure 9.9 shows the two-phase mixture level inside the shroud as a function of core power, for the system pressure of 7.35 MPa, the lower plenum subcooling of 5 K, and the downcomer level changed parametrically. The predicted results for a normalized downcomer level $H_d/H_c$ of 0.77, slightly higher than the top of jet pump, are compared with experimental results of R3-NC-5. The agreement between the predicted and measured mixture levels is very good. It is also seen in this figure that the mixture level inside the shroud rises with core power, however, above a certain power level the mixture level starts to decrease, because the dynamic pressure loss increases rapidly with core power. Similar results are obtained for the system pressure of 2.06 MPa, as shown in Fig. 9.10. The mixture level is higher at 2.06 MPa than at 7.35 MPa due to higher bundle void fraction. The increase of dynamic pressure loss with core power is shown in Fig. 9.11. This figure shows the gravitational head and dynamic pressure loss terms in Eq. (19) versus core power.

Another topic of interest is the core cooling margin under the internal natural circulation conditions. Figure 9.12 shows the minimum critical heat flux (CHF) ratio as a function of core power for the pressure of 7.35 MPa, and the lower plenum subcooling of 5 K. The CHF ratio is defined as the ratio, at a given elevation of core, of CHF devided by the local heat flux. The CHF is calculated by using modified Zuber correlation\(^4\) applicable to low mass flow rates.

$$q_{\text{CHF}} = 0.131 \, h_{fg} \rho_g^{0.3} \left( \frac{g \alpha}{\rho_f + \rho_g} \right)^{0.3} \left( \frac{\rho_f}{\rho_f + \rho_g} \right)^{0.3} (0.96 - \alpha)$$

for $G \leq 100 \, \text{kg/m}^2\cdot\text{s}$ \hfill (20)

The minimum CHF ratios in the experiments are much larger than unity. This is consistent with the experimental observation that no boiling transition (BT) occurred below the core mixture level.

Figure 9.13 shows the mixture level inside the shroud as a function of downcomer water level, for different core powers and system pressures. The present model predicts well the experimental mixture level, except for the encircled data for which the measured mixture levels are higher than predicted, because of higher steam velocity at the upper tieplate due to the flow area reduction.

The above results show that the present analytical model can describe well the thermal-hydraulic phenomena during the BWR internal natural circulation mode.

### 9.6 Applicability of Model to BWR Natural Circulation

The applicability of the present experimental results to actual BWRs is of interest. The
predicted BWR core uncovery criteria, in terms of the downcomer water level, are compared to the ROSA-III core uncovery criteria in Figs. 9.14 and 9.15 for system pressures of 7.35 and 2.06 MPa, respectively. The comparison is made on the core power per unit volume basis.

The ROSA-III and BWR core uncovery criteria agree for core powers less than about 10%. However, for higher core powers the predicted limiting downcomer water level is considerably higher for ROSA-III than for BWR. The difference is larger for the lower pressure (2.06 MPa) than for the higher pressure (7.35 MPa). For higher core powers the BWR core uncovery occurs only after the downcomer-side water level (actually, the water level inside the jet pump diffuser) has decreased much lower than the jet pump suction elevation, whereas the ROSA-III core uncovery initiates for downcomer water levels higher than the jet pump suction. These differences between the predicted BWR and ROSA-III natural circulation behaviors are discussed in the following paragraphs.

ROSA-III simulates the reference BWR, a BWR/6 251-848, with a volumetric scaling factor of 1/424 and a height scaling factor of 1/2. Hence, the core flow area is scaled by a factor of 1/212, however, the flow area restrictions at the core inlet and outlet are volumetrically scaled, by a factor of 1/424, so that the core pressure loss during the nominal operating conditions, with volumetrically scaled core flow rate, will be full scale. The ROSA-III core simulates the cosine axial core power profile of BWR using half-length rods.

The experimental analyses described in the previous sections have shown that the mixture level behavior inside the core shroud is controlled by the static head difference between the core and downcomer, and the dynamic pressure losses in the core and bypass regions. The comparison of the model predictions for BWR and ROSA-III shows that the gravitational head in the ROSA-III core is greater than the height-scaled (1/2-scale) BWR gravitational head, since the core average void fraction is lower than that in BWR. The core void fraction, calculated by Eq. (10), is determined primarily by vapor superficial velocity, \( j_v \), since \( j_e > j_v \). The vapor superficial velocity \( j_e \) is calculated from Eq. (12), as

\[
j_e = \frac{Q_o}{\rho_g h_j A_c} \left[ f_1(z/H_c) - \frac{\Delta h_{s,\ell}}{h_{s,\ell} + 2h_{s,\ell}} \right]
\]

Thus, \( j_e \) at a certain core elevation, \( z/H_c \) is proportional to power per unit core flow area, \( Q_o/A_c \), which is 1/2-scale in ROSA-III when the core power is volumetrically scaled. Considering this, the core uncovery criteria for BWR and ROSA-III, shown in Figs. 9.14 and 9.15, are compared on the basis of power per unit core flow area in Fig. 9.16. Although the agreement between the two criteria is better than in the power-per-volume basis comparisons, for area-scaled core powers smaller than 10%, the difference between the two criteria is large for higher core powers.

The difference between the ROSA-III and BWR limiting downcomer levels for the higher core power levels occurs because of larger-than-scaled dynamic pressure losses in ROSA-III. The primary contributors to the dynamic pressure losses are the area-change losses at flow area restrictions: the single phase pressure loss at the bypass leak holes for the internal circulation flow, and the two-phase pressure loss at the upper tie plate for the primary circulation flow. Since the flow areas at these portions are volumetrically scaled, the dynamic pressure losses are too high to be consistent with the 1/2-scale gravitational heads. The effect is significant for high core volumetric flow rates associated with high core power levels, therefore, higher downcomer water level \( H_c/H_L \) is required in ROSA-III to cover the core by two-phase mixture than in BWR.
9.7 Conclusions

Five natural circulation experiments were conducted in the ROSA-III facility. The conclusions obtained from the analyses of these experiments are:

1) Three distinct natural circulation modes, primary circulation, internal circulation and open loop mode, were identified. The initiation and termination conditions for these modes were experimentally defined for vessel pressures of 7.35 and 2.06 MPa, and for scaled core powers below 20%.

2) An analytical model was developed to simulate the internal (in-shroud) natural circulation mode. This model successfully predicted the in-shroud mixture level for given pressure, core power and downcomer level.

3) For scaled core powers below 20%, critical heat flux did not occur below mixture level. Thus, the core cooling capability is given in terms of core uncovery criterion.

4) The BWR core uncovery criterion, predicted by the present analytical model, agree with that of ROSA-III for core powers less than 10% nominal power.

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### Table 9.1 Test Conditions

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Pressure (MPa)</th>
<th>Power (MW)</th>
<th>Downcomer Level (m)</th>
<th>Lower Plenum Subcooling (K)</th>
</tr>
</thead>
<tbody>
<tr>
<td>R3-NC-1</td>
<td>7.35</td>
<td>1.8 (20%)</td>
<td>5.0 - 2.8</td>
<td>5.0</td>
</tr>
<tr>
<td>R3-NC-2</td>
<td>2.06</td>
<td>1.8 (20%)</td>
<td>5.0 - 2.8</td>
<td>5.0</td>
</tr>
<tr>
<td>R3-NC-3</td>
<td>2.06</td>
<td>0.63 (7%)</td>
<td>5.0 - 2.8</td>
<td>5.0</td>
</tr>
<tr>
<td>R3-NC-4</td>
<td>7.35</td>
<td>0.63 (7%)</td>
<td>5.0 - 2.8</td>
<td>5.0</td>
</tr>
<tr>
<td>R3-NC-5</td>
<td>7.35</td>
<td>1.8 - 0.09 (20% - 1%)</td>
<td>3.07</td>
<td>5.0</td>
</tr>
</tbody>
</table>

*1 Downcomer level was reduced from 5.0 m to 2.8 m.
*2 Core power was reduced from 1.8 MW to 0.09 MW.

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**Fig. 9.1** ROSA-III natural circulation flow path.
Fig. 9.2 Primary circulation flow rate, 7.35 MPa.

Fig. 9.3 Primary circulation flow rate, 2.06 MPa.
Fig. 9.4 Downcomer water level and in-shroud mixture level during R3-NC-1 experiment.

Fig. 9.5 Modes of BWR natural circulation.
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**Fig. 9.7** Primary and internal flow rates, 7.35 MPa.
Fig. 9.8 Primary and internal flow rates, 2.06 MPa.

Fig. 9.9 In-shroud mixture level, 7.35 MPa.
Fig. 9.10 In-shroud mixture level, 2.06 MPa.

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16. Break Location Effects on Thermal-Hydraulics during Intermediate Break

16.1 Introduction

Most investigations in the ROSA-III Experimental Program focused on the main recirculation pump (MRP) suction side pipe break LOCAs, because it has been believed that the LOCAs of this type result in the most severe core dryout and the highest PCT. However, there are many locations where a pipe break may occur besides the MRP suction side pipe. It is important to understand the characteristics of the thermal-hydraulic phenomena of LOCAs caused by a break at various locations in order to verify the safety of a BWR. Therefore, several tests having different break locations were conducted at the ROSA-III facility\(^1\). They include a single-ended jet pump drive (JPD) flow line break test and a main steam line (MSL) break test as well as the MRP suction pipe break test.

In the MRP delivery side pipe break, the rotor of the MRP may be destroyed by excessive fluid flow to the break outlet and the pump rotor may lock. If pump lock occurs, the flow resistance in the pump becomes quite large and flow rate from the pressure vessel to the break outlet through the pump becomes very small. The extreme case of this situation is the single-ended break of the JPD flow line.

In the MSL break experiment, it was assumed that the break occurred between the pressure vessel and the main steam isolation valve (MSIV). In this case, there is no way to stop the steam discharge from the vessel. Thus, it is also a typical LOCA to be investigated in addition to the MRP suction pipe break LOCAs.

16.2 Test Description

Major test conditions of Runs 927, 930, 991 and 992 are briefly described in this section.

Table 16.1 shows the major test conditions. The variable of the test is the break location. In Runs 927 and 930, the break is a split break in the MRP suction line (MRP suction line break : MRPS-B). In Run 991, the break is a single-ended break in the jet pump drive flow line (jet pump drive line break : JPD-B). Thus, the MRP delivery pipe was closed during the transient. The Run 992 is the MSL break (MSL-B) test.

Other test conditions are similar to each other. The effects of the differences in other test conditions are minor in comparing the results of these tests as will be explained.

The break size is 15% for Run 927 (MRPS-B) and Run 992 (MSL-B), 21% for Run 991 (JPD-B) and 25% for Run 930 (MRPS-B). The break size of 100% is defined as the scaled flow area of the MRP suction pipe. Runs 927 and 930 are the base tests for comparison. Runs 927 and 992 are easily compared. The break size of Run 991 is in between those of Runs 927 and 930. Thus, it is possible to compare the JPD-B test with the base MRPS-B tests.

The break size of Run 991 of JPD-B is given by the jet pump drive nozzle size. In contrast, the break sizes of Runs 927, 930 and 992 are given by an orifice or a nozzle in the break unit. It was confirmed in Reference 2 that the difference in break flow rate for the orifice or nozzle is only observed during the subcooled critical flow condition and the effect on the LOCA transient is small. Thus, the effect of the break configuration difference in
Runs 927, 930, 991 and 992 can be neglected.

The initial liquid level in the downcomer is a little different in each experiment. However, the differences of characteristic phenomena observed in the tests are so large, as will be explained in the later, that the effect of the difference in the initial liquid levels can be neglected.

The core power decay curves used in Runs 927 and 930 and Runs 991 and 992 are different. The decay curve used in Runs 927 and 930 is conservative while that used in Runs 991 and 992 is more realistic as discussed in Reference 3. However, according to Reference 3, the effect of the difference in the core power decay curves on the LOCA transient is negligible in intermediate and small break LOCAs, therefore, in a 15% break LOCA.

ECCS conditions are different between tests. A single failure out of three diesel generators for ECCS was assumed for Runs 927 and 930 and no water was injected from the HPCS. In contrast, double failure was assumed in Runs 991 and 992 and no water was injected from the HPCS, and the LPCI had only one third capacity. However, no HPCS injection in common is of importance. The effect of HPCS failure is most severe among the three possible single failure in the ROSA-III tests, as described in the preceding chapter. Since the most interesting and characteristic phenomena were observed during the blowdown phase before the initiation of LPCS, as will be discussed in the following section, the differences in the LPCS and LPCI conditions are not important. In a steam line break LOCA situation, the high pressure signal in the containment building may trip the ECCS. This high pressure signal activation was also assumed to fail in Run 992.

Liquid level signals in the downcomer were used to initiate MSIV closure and ECCS actuation in the tests. The liquid level setting is a little different in each experiment. However, the differences in important and interesting phenomena observed were not caused by that difference as will be explained later. Thus, the liquid level setting difference is considered negligible and not important.

The primary initial test conditions before the break were as follows. The steam dome pressure was 7.35 MPa and the corresponding saturation temperature was 562 K. The core inlet flow rate was 16 kg/s and the core outlet quality was 14%. The lower plenum subcooling was 11 K.

16.3 Test Results and Discussions

The test results are presented in this section and the characteristic phenomena are discussed.

The chronologies of major events for the four tests are summarized in Table 16.2.

16.3.1 Pressure Transients

The measured vessel pressure transients for the four tests are compared in Fig. 16.1. The pressure transients of the 15% and 25% MRPS-B tests and the 21% JPD-B test are similar. However, the pressure transient of the MSL-B test is quite different from that of the other three tests.

In the two MRPS-B tests and the JPD-B test, the vessel pressure began to decrease immediately after break initiation. When the MSIV closed, the vessel pressure started to increase in the three tests. The vessel pressure increasing rates in the two MRPS-B tests were larger than that in the JPD-B test. This was caused by the different core power decay curves used. Generally, the core power decay curve used in the ROSA-III test simulated the decay power of the fission products and actinides, the delayed neutron fission power and the stored
heat release from the nuclear fuel rod\textsuperscript{39}. The core power decay curve used in the MRPS-B tests was a little higher than that used in the JPD-B test because the larger stored heat was assumed in the former than in the latter where it was estimated realistically. Thus, the heat generation in the core in the MRPS-B tests was a little larger than that in the JPD-B test during the early stage of blowdown. The difference between the two power curves decreased with time and became negligible by 100 s. Thus, the vessel pressure increase after the MSIV closure was large in the MRPS-B tests and small in the JPD-B test. It can be concluded that the observed differences in the pressure increase after the MSIV closure between the MRPS-B and JPD-B tests is not because of different break locations and is not important.

The vessel pressure increase gradually slowed down and finally reversed because of core power decay in the two MRPS-B tests. The depressurization increased after the recirculation line uncovering (RLU) and start of core dryout. The RLU is defined as the uncovering of the outlet nozzle to the MRP in the downcomer. Thus, after the RLU, steam in the vessel could exit the vessel and depressurization was accelerated. When the core dryout began, steam generation in the core decreased, which resulted in further depressurization. In contrast, depressurization was mainly accelerated by the initiation of core dryout in the JPD-B test. Thus, the depressurization rate until ADS actuation in the 21\% JPD-B test was slower than those in the 15\% and 25\% MRPS-B tests. The water level transient in the downcomer and the core dryout will be discussed further in the later sections.

After the ADS actuation, the depressurization was accelerated in the JPD-B test because of discharge initiation of steam from the ADS and the depressurization rate became almost the same as that in the 25\% MRPS-B test.

After the ADS actuation, the LPCS and then the LPCI began to inject water into the vessel in the MRPS-B tests and the LPCI in the JPD-B test when the system pressure reached 2.2 and 1.6 MPa, respectively.

In the MSL-B test, the vessel pressure decreased gradually after the break. The depressurization was caused by discharge of vessel steam from the break. ADS actuation later in the transient slightly increased the rate of depressurization as a result of steam discharge from the ADS. After the ADS actuation, the LPCI began to inject water when the vessel pressure reached 1.6 MPa.

\subsection{16.3.2 Water Level Transients}

Water level transients in the downcomer, the lower plenum, the core and the upper plenum measured in the four tests are shown in Fig. 16.2 through 16.5. The water levels shown in the figures were measured with the conductance probes. Thus, the water levels in the downcomer and the lower plenum shown in the figures are single-phase liquid levels and two-phase mixture levels before and after initiation of flashing, respectively. The water levels in the core and the upper plenum are two-phase mixture levels throughout the transients.

As for the vessel pressure transients, the water level transient in the MSL-B test is quite different from those in the MRPS-B and JPD-B tests. Even between the MRPS-B and JPD-B tests large differences are observed in the water level transients.

In the 15\% and 25\% MRPS-B tests, the downcomer water level began to decrease after the break and reached the recirculation line outlet. The inside shroud mixture level began to decrease in the upper plenum first, and dropped into the core. When the lower plenum pressure reached the saturation pressure of 6.4 MPa due to the vessel depressurization, lower plenum flashing (LPF) occurred. In the 25\% MRPS-B test, the core mixture level swelled because of the LPF initiation and the core was filled with the two-phase mixture temporarily. In the 15\% MRPS-B experiment, core mixture level swell due to the LPF initiation was not
observed, because the break size was so small that the depressurization rate was not enough to swell the core mixture level. Later in the 15% MRPS-B test, there was a large core mixture level swell and the core was filled with two-phase mixture temporarily when the ADS was actuated. Small core mixture level swell was also observed in the 25% MRPS-B test when the ADS started. The lower plenum pressure was 6.13 and 4.75 MPa in the 15% and 25% MRPS-B tests, respectively when the ADS was activated. Thus, impact of the ADS actuation on the LPF was larger in the 15% MRPS-B test than in the 25% MRPS-B test. In both tests, the lower plenum mixture level started to recover after the LPCS initiation and the core mixture level was restored quickly after the LPCI initiation.

An interesting phenomenon in the MRPS-B tests is that the core mixture level started to drop after the RLU which resulted in steam discharge from the vessel through the recirculation line outlet nozzle and rapid depressurization.

In the JPD-B test, the downcomer water level started to drop after the break. However, it stopped dropping when it reached the jet pump suction elevation and stayed there until the ADS actuation. The core mixture level fell quicker than in the 15% and 25% MRPS-B tests and the core had dried out before the ADS was actuated. After the ADS actuation, the downcomer mixture level recovered temporarily and started again to drop, however, the rate of decrease was much less than that just after the break initiation.

Figure 16.6 illustrates why the downcomer and core water level transients were different between the MRPS-B and JPD-B tests.

In the MRPS-B tests, steam in the vessel began to flow out of the break when the downcomer water level dropped to the recirculation line outlet nozzle elevation. Therefore, coolant loss in the core is reduced and the depressurization was accelerated. In the JPD-B test, steam in the vessel could not flow out freely from the break even when the downcomer water level dropped to the jet pump suction elevation since the jet pump drive nozzle elevation was lower than the jet pump suction elevation and the drive nozzle was submerged in water. Thus, coolant in the core continued to flow out through the jet pump and the jet pump drive nozzle in a reverse flow direction. Accordingly, the downcomer water level stayed at the jet pump suction elevation, the core mixture level continued to drop until the ADS actuation and the depressurization was slower than in the MRPS-B tests.

When the ADS started to discharge steam from the vessel, rapid depressurization started and flashing in the vessel became significant in the JPD-B test. Thus, the downcomer mixture level swelled temporarily by flashing and began to drop due to mass depletion by the flashing. The fluid quality at the jet pump drive nozzle became high and the discharge flow rate through the jet pump drive nozzle decreased. However, the core mixture level continued to drop after temporary recovery since steam discharge through the ADS was added and vaporization of fluid in the core continued. After the start of the LPCI injection, the core and downcomer mixture levels were restored.

In the MSL-B test, the break hole is at the top of the vessel. Thus, discharge flow was steam or high quality flow throughout the whole transient. The coolant loss occurred only by vaporization. Thus, the coolant loss was slower than in the 15% MRPS-B test, as shown in Fig. 16.5. The depressurization was also slower than in the 15% MRPS-B test even before the ADS actuation since more fluid remained in the vessel in the 15% MSL-B test than in the 15% MRPS-B test. The downcomer was filled with a two-phase mixture throughout the transient, and the collapsed level dropped slower than in the 15% MRPS-B test, as shown in Fig. 16.7. Thus, trip signal (L1 level signal) for the ADS actuation and the LPCI injection initiation was activated much later than in the 15% MRPS-B test. The slower decrease in the downcomer collapsed level coupled with the slower depressurization to cause much later
initiation of the LPCI than in the 15% MRPS-B test. The core mixture level drop was much later than that in the 15% MRPS-B test as shown in Fig. 16.2 and 16.5. However, the delayed initiation of the LPCI resulted in a long core boil-off period. Thus, two-thirds of the core dried. After the LPCI initiation, the core mixture level recovered quickly.

16.3.3 Cladding Temperature Transients

Cladding temperatures measured in the 15% and 25% MRPS-B tests, the 21% JPD-B test and the 15% MSL-B test are shown in Fig. 16.8. The cladding temperatures shown were measured at the top, middle and bottom of the A-11 rod which was located at the corner of the high power bundle and had the highest local peaking factor. The cladding temperature transients measured on other rods were similar to that of the A-11 rod in each experiment.

The initiation of cladding surface temperature excursion in the 21% JPD-B test was earlier and the peak temperatures were higher than those in the 15% and even 25% MRPS-B tests. This can be inferred from the core mixture level transients shown in Figs. 16.3, 16.4 and 16.5.

There is little value in comparing the peak cladding temperature (PCT) in each experiment since the ECCS actuation specification is different in each case. However, in each experiment, the measured PCT was much lower than the present LWR safety criterion of 1473 K.

16.3.4 Discussions

The characteristic phenomenon observed in the JPD-B test was quick core mixture level depression. The jet pump nozzle was submerged in low quality fluid until the ADS actuation and steam generated in the vessel could not get to the jet pump drive nozzle and depressed the core mixture level. Thus, low quality fluid discharge continues for a long time until the ADS actuation. The JPD-B has the possibility of resulting in more severe core dryout than the MRPS-B. It should be noted that the maximum size of the JPD-B is limited to 21% because the total flow area of the jet pump drive nozzles in one recirculation loop is 21%.

Characteristic phenomena observed in the MSL-B experiment were two-phase mixture level swell in the downcomer and the core for a long time and slow coolant loss. Since the fluid exit was at the top of the vessel, only steam or high quality fluid could leave the vessel. The two-phase mixture level swell and slow coolant loss in the core kept the core covered with the two-phase mixture for a long time and delayed the core dryout. However, the slow coolant loss in the downcomer delayed the ECCS trip signal activation and resulted in core boil-off initiation and cladding temperature excursion. Thus, if other ECCS trip signals fail and the downcomer water level is the only signal available to activate coolant injection into the vessel and the HPCS fails to inject coolant, core dryout and cladding temperature excursion may occur during a MSL-B LOCA of a BWR. However, the ROSA-III test results infer that the core dryout and the cladding temperature excursion in the MSL-B LOCA might be milder than those in the MRPS-B LOCA since the core dryout occurs much later and heat generation in the core is lower during the dryout period than in the MRPS-B LOCA.

16.4 Conclusions

Break location effects on thermal-hydraulics during intermediate break size LOCAs in a BWR were investigated by using the results of four tests at the ROSA-III facility, the 15% and 25% main recirculation pump suction pipe break (MRPS-B) tests, the 21% single-ended jet pump drive flow line break (JPD-B) test and the 15% main steam line break (MSL-B) test. No
water was injected from the HPCS since it was assumed to be unavailable in all the experiments. The following conclusions were obtained:

1. In the MRPS-B LOCA, when the water level in the downcomer dropped to the recirculation line outlet elevation, the discharge flow changed from low quality to high quality fluid flow. It resulted in slow coolant loss from the vessel and the core, and rapid depressurization.

2. In the JPD-B LOCA, the jet pump drive nozzle was covered with low quality fluid until the ADS actuation even after the water level in the downcomer dropped below the jet pump suction elevation. Thus, steam generated in the vessel could not flow out of the vessel and the core mixture level was depressed. Low quality fluid discharge through the jet pump drive nozzle continued and the core dried out before the ADS actuation. The core dryout was faster and earlier than those in the 15% and even 25% MRPS-B LOCAs. The JPD-B LOCA has the possibility to cause severer core dryout and cladding temperature excursions than the MRPS-B LOCA.

3. The MSL-B LOCA was characterized by mixture level swell in the downcomer and the core. The core mixture level swell resulted in much later core dryout initiation than that in the MRPS-B LOCA. However, the downcomer collapsed liquid level drop was also much later since the level drop was caused only by vaporization. This resulted in later ECCS actuation since it was assumed that the high containment pressure failed to trip the signal for ECCS actuation. While the ECCS actuation might be delayed in the MSL-B LOCA, the resulting core dryout may not be so severe and the cladding temperature excursion may be lower than in a MRPS-B LOCA since the heat generation rate in the core is very low when the core dryout occurs.

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## Table 16.1 Experimental conditions

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Run 927 (MRPS-B)(^{(1)})</th>
<th>Run 930 (MRPS-B)(^{(1)})</th>
<th>Run 991 (JPD-B)(^{(2)})</th>
<th>Run 992 (MSL-B)(^{(3)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break Location</td>
<td>MRP suction pipe</td>
<td>MRP suction pipe</td>
<td>JP drive flow pipe</td>
<td>MSL pipe</td>
</tr>
<tr>
<td>Break Size (%)</td>
<td>15</td>
<td>25</td>
<td>21</td>
<td>15</td>
</tr>
<tr>
<td>Break Configuration</td>
<td>Nozzle</td>
<td>Orifice</td>
<td>Nozzle</td>
<td>Orifice</td>
</tr>
<tr>
<td>Initial Core Power (MW)</td>
<td>3.97</td>
<td>3.96</td>
<td>3.96</td>
<td>3.97</td>
</tr>
<tr>
<td>Core Power Decay Curve(^{(4)})</td>
<td>Conservative</td>
<td>Conservative</td>
<td>Realistic</td>
<td>Realistic</td>
</tr>
<tr>
<td>Initial Water Level (m)</td>
<td>5.03</td>
<td>5.04</td>
<td>4.80</td>
<td>4.81</td>
</tr>
<tr>
<td>MSL Closure</td>
<td>L2(^{(5)}) + 3 s</td>
<td>L2(^{(5)}) + 3 s</td>
<td>L1(^{(6)}) + 3 s</td>
<td>L1(^{(6)}) + 3 s</td>
</tr>
<tr>
<td>ECCS</td>
<td>ADS (L1(^{(6)}) + 120 s)</td>
<td>ADS (L1(^{(6)}) + 120 s)</td>
<td>ADS (L1(^{(6)}) + 120 s)</td>
<td>ADS (L1(^{(6)}) + 120 s)</td>
</tr>
<tr>
<td>LPCS</td>
<td>LPCS</td>
<td>No LPCS</td>
<td>No LPCS</td>
<td>No LPCS</td>
</tr>
<tr>
<td>(L1(^{(6)}) + 40 s + 2.2 MPa)</td>
<td>(L1(^{(6)}) + 40 s + 2.2 MPa)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3LPCI</td>
<td>3LPCI</td>
<td>2LPCI</td>
<td>2LPCI</td>
<td></td>
</tr>
<tr>
<td>(L1(^{(6)}) + 40 s + 1.6 MPa)</td>
<td>(L1(^{(6)}) + 40 s + 1.6 MPa)</td>
<td>(L1(^{(6)}) + 40 s + 1.6 MPa)</td>
<td>(L1(^{(6)}) + 40 s + 1.6 MPa)</td>
<td></td>
</tr>
</tbody>
</table>

**Note:**

1. MRPS-B: Main Recirculation Pump Suction Line Break
2. JPD-B: Jet Pump Drive Line Break
3. MSL-B: Main Steam Line Break
4. Please refer to Reference 3 and Chapter 10.
5. L2 in Runs 927 and 930 is 4.76 m.
6. L1 in Runs 927 and 930 is 4.00 m. L1 in Runs 991 and 992 is 4.25 m.
### Table 16.2: Chronology of major events

#### (a) Run 927, 15% MRPS-B test

<table>
<thead>
<tr>
<th>Time (s)</th>
<th>Events</th>
<th>Note:</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>Break started</td>
<td>(1) Main Steam Isolation Valve: MSIV</td>
</tr>
<tr>
<td>4</td>
<td>Feedwater stopped</td>
<td>(2) Recirculation Line Uncovery: RLU</td>
</tr>
<tr>
<td>9</td>
<td>Core power decay started</td>
<td>(3) Lower Plenum Flashing: LPF</td>
</tr>
<tr>
<td>17</td>
<td>L2 level</td>
<td>(4) Automatic Depressurization System: ADS</td>
</tr>
<tr>
<td>25</td>
<td>MSIV closed</td>
<td>(5) Low Pressure Core Spray: LPCS</td>
</tr>
<tr>
<td>32</td>
<td>L1 level</td>
<td>(6) Low Pressure Coolant Injection: LPCI</td>
</tr>
<tr>
<td>79</td>
<td>Downcomer mixture level dropped to recirculation line outlet (RLU)</td>
<td></td>
</tr>
<tr>
<td>117</td>
<td>LPF(2) started</td>
<td></td>
</tr>
<tr>
<td>153</td>
<td>ADS(4) was actuated</td>
<td></td>
</tr>
<tr>
<td>283</td>
<td>LPCS(5) started</td>
<td></td>
</tr>
<tr>
<td>350</td>
<td>LPCI(6) started</td>
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</tr>
</tbody>
</table>

#### (b) Run 930, 25% MRPS-B test

<table>
<thead>
<tr>
<th>Time (s)</th>
<th>Events</th>
</tr>
</thead>
<tbody>
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<td>Break started</td>
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<tr>
<td>3</td>
<td>Feedwater stopped</td>
</tr>
<tr>
<td>9</td>
<td>Core power decay started</td>
</tr>
<tr>
<td>11</td>
<td>L2 level</td>
</tr>
<tr>
<td>16</td>
<td>MSIV closed</td>
</tr>
<tr>
<td>18</td>
<td>L1 level</td>
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<tr>
<td>32</td>
<td>Downcomer mixture level dropped to recirculation line outlet (RLU)</td>
</tr>
<tr>
<td>73</td>
<td>LPF started</td>
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<tr>
<td>139</td>
<td>ADS was actuated</td>
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<tr>
<td>217</td>
<td>LPCS started</td>
</tr>
<tr>
<td>272</td>
<td>LPCI started</td>
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</table>

#### (c) Run 991, 21% JPD-B test

<table>
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<th>Time (s)</th>
<th>Events</th>
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<tbody>
<tr>
<td>0</td>
<td>Break started</td>
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<tr>
<td>4</td>
<td>Feedwater stopped</td>
</tr>
<tr>
<td>8</td>
<td>Core power decay started</td>
</tr>
<tr>
<td>17</td>
<td>L2 level</td>
</tr>
<tr>
<td>26</td>
<td>L1 level</td>
</tr>
<tr>
<td>29</td>
<td>MSIV closed</td>
</tr>
<tr>
<td>45</td>
<td>Downcomer mixture level dropped to JP suction elevation</td>
</tr>
<tr>
<td>148</td>
<td>LPF started</td>
</tr>
<tr>
<td>168</td>
<td>ADS was actuated</td>
</tr>
<tr>
<td>266</td>
<td>LPCI started</td>
</tr>
</tbody>
</table>

#### (d) Run 992, 15% MSL-B test

<table>
<thead>
<tr>
<th>Time (s)</th>
<th>Events</th>
</tr>
</thead>
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<td>Break started</td>
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<td>4</td>
<td>Feedwater stopped</td>
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<td>8</td>
<td>Core power decay started</td>
</tr>
<tr>
<td>12</td>
<td>MSIV closed</td>
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<tr>
<td>23</td>
<td>LPF started</td>
</tr>
<tr>
<td>167</td>
<td>L2 level</td>
</tr>
<tr>
<td>445</td>
<td>L1 level</td>
</tr>
<tr>
<td>592</td>
<td>ADS was actuated</td>
</tr>
<tr>
<td>730</td>
<td>LPCI started</td>
</tr>
</tbody>
</table>
Fig. 16.1 Vessel pressure transients in 15% and 25% MRPS-B tests, 21% JPDB test and 15% MSL-B test.

Fig. 16.2 Water level transients in vessel in 15% MRPS-B test.
Fig. 16.3  Water level transients in vessel in 25% MRPS-B test.

Fig. 16.4  Water level transients in vessel in 21% JPD-B test.
Fig. 16.5 Water level transients in vessel in 15% MSL-B test.

Fig. 16.6 Illustration of flow path in MRPS-B and JPD-B tests.
Fig. 16.7  Downcomer collapsed level transients in 15% MRPS-B and MSL-B tests.

Fig. 16.8  Cladding temperature transients of A-11 rod in 15% and 25% MRPS-B test, 21% JPD-B test and 15% MSL-B test.
10. Power Curve Sensitivity Test Series

10.1 Introduction

In the ROSA-III facility, the nuclear fuel rods are simulated by electrically heated rods. As noted in Reference 1, simulation of the power generation in ROSA-III tests after scram has a large effect on the system transient. The simulation of the power generation is complicated by the physical property difference between uranium dioxide (UO₂) in the nuclear fuel rod and insulator material in the heater rod. Also, in the nuclear fuel rod, the power generation is distributed in the fuel pellet, but in the electrically heated rod the heat generation is concentrated in the heater element. Thus, the derived heater rod power curve for the ROSA-III tests was based on the heat release from a nuclear fuel rod to coolant rather than the fuel rod power generation history²). However, because the effect of the power generation rate on the system transient, especially on the heater rod surface temperature, was expected to be large, the influence of the power curve assumption was examined using a series of heater rod power curves in the ROSA-III tests.

The test series consisted of seven experiments and the break area was changed in the range of 5 to 200%. The break location was at the recirculation pump inlet line. Three kinds of the power input curve to heater rods were used; base case with conservative evaluation of initial stored heat, a case with realistic initial stored heat and a case with no initial stored heat. In the series, a single failure of HPCS diesel generator was assumed because the maximum PCT in each break size had been obtained when the failure of the HPCS diesel generator was assumed in the ROSA-III tests.

10.2 Test Description

10.2.1 Power Curves

After a reactor is scrammed, the reactor power is composed of three terms, the delayed neutron fission power, the decay power of the fission products and actinides, and the stored heat release from the fuel rod. As noted in References 1 and 4, a major portion of the initial stored heat in the fuel rod is released in the first 30 s after scram in a large break LOCA. Figure 10.1 shows the comparison of the calculated total heat transfer rate to coolant and the stored heat release term during a double-ended break LOCA of a BWR. RELAPS/Mod1/CY1⁵ was used for the calculation. Details are included in Reference 4. The calculated stored heat release term for a ROSA-III heater rod is also shown in the figure. In the ROSA-III calculation the heat transfer rate to coolant of the BWR is simulated. In the first 30 s the stored heat release term is the main contributor to the heat transfer to coolant and is much larger than other two terms; i.e., fission and decay power.

The heat capacitance of the boron nitride (BN) insulator in a ROSA-III heater rod is very close to that of uranium dioxide (UO₂). However, the thermal conductivity of BN is larger than that of UO₂ by a factor of ten. In the nuclear fuel rod the heat generation is distributed in the fuel pellet, but, in the heater rod the heat generation is concentrated in the microme. Thus the center line temperature of the nuclear fuel is much higher than that of the heater rod and the initial stored heat in the heater rod is very small compared with that of the nuclear fuel rod. The power curve after initiation of the experiment has to be carefully designed to
produce the same heat transfer rate to the coolant as the nuclear fuel rod.

Three power curves were prepared for ROSA-III power curve sensitivity test series based on RELAP4J\(^6\) and RELAP5/Mod1/CY1 code calculations for a BWR; Base Curve, Realistic Curve and ANS Curve. These curve are shown in Fig. 10.2. Each curve has a constant part after initiation of the transient because of the limited capacity of the ROSA-III power supply system.

(1) Base Curve

There are uncertainties in physical properties of \(\text{UO}_2\) and gas in the fuel rod, and gap width between the pellet and cladding. Because the ROSA-III tests were to produce conservative results, the heat capacitance of \(\text{UO}_2\) was multiplied by a factor of 1.5 to obtain conservative results of the stored heat. In order to quantify the heat transfer rate from a nuclear fuel to the coolant, an analysis of a BWR double-ended break LOCA\(^2\) was conducted by RELAP4J. Delayed neutron fission power, the decay power of the fission products and actinides, and the stored heat release were taken into consideration and reactivity feedbacks of moderator density change and Doppler effect were evaluated properly. A power curve for ROSA-III heater rods was derived from the calculated heat transfer rate from the nuclear fuel rod to coolant. On that stage, the heat capacitance of the ROSA-III heater rod was neglected for the sake of conservatism. After 260 s, the power curve was set to be equal to ANS decay power curve\(^7\) because the difference between the calculated power curve and the ANS decay power curve became little.

(2) Realistic Curve

To get realistic correspondence of thermal-hydraulic response between a BWR and the ROSA-III, the Realistic Curve for power input to ROSA-III heater rods was developed. RELAP5/Mod1/CY1 was used for the analysis to get the curve. Realistic thermal properties of \(\text{UO}_2\) and gas, especially heat capacitance and thermal conductivity of \(\text{UO}_2\), were used in the calculation. Heat transfer rate from a nuclear fuel rod to coolant during a double-ended break LOCA transient of a BWR was calculated by RELAP5/Mod1/CY1 and was converted to the power curve for ROSA-III heater rods. On that stage, heat capacitance of the heater rod was properly taken into consideration. After 200 s, the power curve was set to be equal to ANS decay power curve because the difference between the calculated power curve and the ANS decay power curve became little. The details are described in Reference 4.

(3) ANS Curve

The Japanese guideline\(^8\) for LWR licensing allows to use the ANS decay power curve\(^7\) to calculate the heat generation rate in a nuclear fuel after scram in safety analysis of a nuclear power plant to determine ECCS effectiveness in a LWR. Initial stored heat in a nuclear fuel rod is important in the early portion of a LOCA transient as discussed already. If the ANS decay power curve is used in a ROSA-III test as a power curve for heater rods, the heat transfer rate to coolant would be much lower than that of a nuclear fuel rod of a BWR in the early portion of the LOCA transient. The power curve of ANS decay power curve gives the lower bound of the heat transfer rate to coolant because the stored heat is not taken into account.

10.2.2 Test Conditions

The test conditions are summarized in Table 10.1 for the power curve sensitivity test series. In each experiment the failure of the HPCS diesel generator was assumed and no ECCS water was injected from HPCS. The break location was at the recirculation pump inlet line and the break type was 200% double-ended break in Runs 906, 907, and 926, 50% split break in Runs 916 and 933, and 5% split break in Runs 916 and 934. Blowdown was initiated by opening the quick opening blowdown valve(s) at the immediate downstream of the break
orifices or nozzles.

The primary test initial conditions before break were as follows. The steam dome pressure was 7.35 MPa and the corresponding saturation temperature was 562 K. The steady state power was 3.96 MW corresponding to the maximum linear heat rate (MLHR) of 16.7 kW/m. The core inlet flow rate was 16 kg/s and the core outlet quality was 14%. The lower plenum subcooling was 11 K.

Test procedures used were standard ones as described in Sec. 2.5.

The power curves of Base and Realistic Curves were used in the experiments irrespective of break size. Fundamentally, the heat transfer rate to coolant during a BWR LOCA depends on the flow condition, which means the power input curves to heater rods based on the simulation of the heat transfer rate to coolant should be changed with the break size. However, the effect of the break size on the power curve is negligible as is discussed in Reference 3.

10.3 Test Results

10.3.1 200% Double-Ended Break

The system pressure transients in Runs 926, 907 and 906 are shown in Fig. 10.3. There are small differences in the system pressure transients. The principal trends are summarized in Table 10.2. The system pressure in each case decreases immediately after the break as fluid is discharged. However, the pressure starts to increase after the main steam isolation valve (MSIV) closure at 6 s during Run 926 as the steam volumetric discharge flow rate from the break becomes less than the steam volumetric generation rate in the core. The mixture level in the downcomer decreases rapidly after the break and reaches the recirculation line outlet nozzle (RLU, recirculation line recovery) at 13 s. Thereafter, the vessel steam begins to exit through the vessel side break and the system pressure decreases rapidly. Similar behavior occurs in Run 907 (Realistic Curve) after the MSIV closure at 10 s. However, the pressure recovery is less than that in Run 926 because the vapor generation rate is smaller than that in Run 926 because of lower power input than Run 926. In the lowest power input case (ANS Curve, Run 906), the pressure never recovers even after the MSIV closure at 7.4 s.

The lower plenum fluid saturates at 17 s, 17 s and 16 s after the break in Runs 926, 907 and 906, respectively, and the lower plenum fluid begins flashing. The system depressurization rate decreases after the initiation of lower plenum flashing (LPF) because of the continuous steam generation in the lower plenum. In each transient, the fluid in the feedwater line saturates at the system pressure of 2.14 MPa. Thus, the system depressurization rate is further decreased as steam flows from the feedwater line to pressure vessel. LPCS and LPCI are actuated at system pressures of 2.1 MPa and 1.5 MPa, respectively.

The differences between the pressure transients of Runs 926, 907 and 906 are small. The break flow differences are also small as shown in Fig. 10.4 because the break size is the same in these runs. Thus, the L2 level trip for MSIV closure is activated at nearly the same time in these runs and the LPCS and LPCI systems initiate to inject cooling water at nearly the same time.

The mixture level transient in the pressure vessel in Runs 926, 907 and 906 are compared in Fig. 10.5. The mixture level is measured by conductivity probes and defined herein as the interface between the steam region and the steam and water mixture region. The mixture level behavior is important in a BWR LOCA because it dominates the rod surface temperature excursion. The core mixture level begins to decrease at 10 s after the break in Run 926 and the uncovering of the upper part of the core occurs before the initiation of the LPF. The same phenomena occurs in Run 907. However, in Run 906, no core uncovering occurs before the
LPF. The larger heat transfer rates (power inputs to heater rods) in Runs 926 and 907 than that in Run 906 result in larger core vaporization rates than that in Run 906, which causes core uncoverage before the initiation of the LPF. After the initiation of the LPF, the core is filled again with the two phase mixture in Runs 926 and 907. The core uncoverage is initiated again at 39 s after the break and the core is completely uncovered by 69 s in Run 926. In Run 907, core uncoverage begins at 56 s and is completed at 74 s. In Run 906 the core uncoverage begins at 61 s, but the complete uncoverage does not occur. Mixture level restoration in the core in Run 906 occurs quickly after LPCI injection initiation and is completed at 110 s. In Run 907 the core mixture level begins to recover at 99 s as the LPCI system initiated to inject coolant. The core is filled with a two-phase mixture by 116 s. In Run 926 the core mixture level begins to recover at 104 s and the core is filled with a two phase mixture by 136 s.

Some differences are observed in the core mixture level transients of Runs 926, 907 and 906. However, the sequence of events is the same. Differences between the transients are caused primarily by the differences in the power curves. The higher power generation in Run 926 than in Run 907 results in higher core liquid vaporization. Thus, the core mixture level decreases earlier in Run 926 than in Run 907. On the contrary, the core mixture level decreases later in Run 906 than in Run 907 because of lower power input in Run 906 than in Run 907.

The core mixture level behavior after LPCI actuation in each run is consistent with the power curve. In each case, the core mixture level is restored by the LPCI system. LPCI is actuated at nearly the same time in each run. The vessel liquid inventory at LPCI initiation in Run 906 is larger than in Run 907 because of the lower power input and less vaporization. Thus, the core mixture level restoration in Run 906 is earlier than that in Run 907. Core mixture level restoration in Run 926 is the latest among the three transients since the base power curve results in maximum vessel liquid inventory vaporization and minimum residual liquid.

The lower plenum mixture level shows basically similar behavior in the three runs. The mixture level begins to decrease after initiation of the LPF and begins to increase after LPCS initiation. The lower plenum mixture level temporarily shrinks after initiation of LPCI injection because of vapor condensation in the mixture by the cold LPCI water. The lower plenum mixture level behavior is mainly dominated by the LPF and by the water accumulation from the LPCS and LPCI systems.

The mixture level transients in the downcomer in Runs 926, 907 and 906 are almost identical with each other. The downcomer liquid level decreases as the liquid is discharged from the break. The level is restored after the core is reflooded. The downcomer level transient is controlled primarily by the break flow and ECCS flow rate which are similar in all three transients.

The surface temperatures of the heater rod A-11 in Runs 926, 907 and 906 are compared in Fig. 10.6. The surface temperature is measured at seven elevations in the core as shown. The rod A-11 is the peak power rod with a local peaking factor of 1.1 at the corner of the peak power bundle A. The trends of the core mixture levels are also shown in the figure.

The surface temperature begins to increase rapidly between 5 and 10 s in Run 926 as boiling transition occurs above the core midplane. The surface temperature exhibits rewetting between 17 and 35 s as liquid surges into the core during the LPF and core cooling is improved. In Run 907 the surface temperature excursion before the LPF is not prominent. No boiling transition is observed before the LPF in Run 906. In this period differences of power input to heater rods among Runs 926, 907 and 906 are large and the differences are
responsible for the core temperature differences.

As the LPF becomes less violent, the mixture level in the core starts to fall and the fuel surface temperature begins to rise, first at the top of the core and then at the lower elevation. A strong correlation exists between the fuel surface temperature behavior and the mixture level behavior in the core as shown in Fig. 10.6. The initiation time of the fuel surface temperature excursion exactly corresponds to the uncovering time of the fuel surface at the location. The temperature rise at the top of the core in Runs 907 and 906 is not significant because the LPCS spray injection in the upper plenum is initiated immediately after heater rods begins to dry out.

After initiation of the LPCS in Run 926 the fuel surface rewetting occurs temporarily in the upper part of the core. Also heat transfer is improved in the lower portion of the core as water passes from the upper plenum into the core. The core is quenched from the bottom of the core by reflooding after the LPCI initiation in Run 926. However, quenching is propagated from the top to the bottom of the core after the LPCS initiation in Runs 907 and 906, since heater rod surface temperatures are not very high. Even so, core mixture level restoration does not occur until LPCI initiation in Runs 907 and 906.

The slope of the surface temperature rise after mitigation of the LPF and the maximum temperature at the same elevation is highest in Run 926, medium in Run 907 and lowest in Run 906 among three, which is well explained by the differences of power curves or initial stored heats.

10.3.2 50% Break

Pressure transients of Run 916 and Run 933 are compared in Fig. 10.7. The basic trends are the same in both runs. The principal sequence of events is summarized in Table 10.3. The sequence of events is the same as in the 200% double-ended break tests of Runs 926, 907 and 906; pressure decreases immediately after break, pressure recovers after the MSIV closure, the LPF occurs and the LPCS and the LPCI start. The pressure recovery after the MSIV closure in Run 933 is less than that of Run 916 because the power input to heater rods is lower. The pressure in Run 933 is a little lower than that of Run 916 during the transient, however, the difference between the Base Curve and Realistic Curve has minor effect on the pressure transients in the 50% LOCAs.

Mixture level transients in the pressure vessel in Runs 916 and 933 are compared in Fig. 10.8. There is no major difference between the mixture level transients of Runs 916 and 933. The mixture level transient in the core in Run 916 is fundamentally the same as that of a 200% double-ended break experiment of Run 926; temporary recovery of two phase mixture because of the initiation of the LPF, completion of core uncovering and the mixture level restoration after the LPCI actuation.

In Run 933 the core mixture level transient is the same as that of Run 916. The power input to heater rods in Run 933 is lower than the power input in Run 916 and it is suspected that the core mixture level decreases in Run 933 is later than that of Run 916 because of a lower core steaming rate. The pressure in Run 933 is a little lower than the pressure in Run 916 as shown in Fig. 10.7. The lower pressure results in the earlier LPF in Run 933 than in Run 916. The earlier initiation of the LPF in Run 933 compensates the less vaporization in the core. Thus the core mixture level decrease in Run 933 is the same as that of Run 916. The core mixture level recovery is earlier than the recovery in Run 916 by 19 s because the system pressure is a little lower in Run 933 than Run 916, thus the LPCI is actuated earlier.

The lower plenum mixture level begins to decrease before the core dries in Runs 916
and 933. The lower plenum mixture levels begin to recover after LPCS initiation. The lower plenum is filled up again after the core reflooding.

The downcomer is dried out due to coolant loss from break in Runs 916 and 933 and filled again because of accumulation of LPCS and LPCI water.

The surface temperatures of the simulated fuel rod A-11 in Run 916 and Run 933 are compared in Fig. 10.9. The trends of the core mixture levels are also shown. The surface temperature transients have very good correspondence to the core mixture level transients in both cases as was discussed in 200% double-ended break tests. Temporary surface temperature rise is observed at the upper part of the core before the LPF in Run 916 but not in Run 933.

As the LPF subsides, the fuel surface temperatures begin to increase from the top to the bottom of the core in both cases. Quenching propagates from the top of the core after the LPCS initiation. Bottom-up quenching also occurs as LPCS water accumulates in the core. This is accelerated by LPCI initiation. The dryout occurs at the same time and at the same elevation in both cases. The temperature increasing rate after the dryout at the same elevation in Run 916 is larger than in Run 933 and the quenching is later.

10.3.3 5% Break

System pressures of Runs 922 and 934 are compared in Fig. 10.10. The principal sequence of events is summarized in Table 10.4.

The pressure begins to decrease in Run 922 immediately after break. Depressurization is accelerated from 9 s as the core power begins to decrease at 9 s. As the MSIV begins to close at 25 s, the pressure begins to increase. However, the pressure is maintained below 8.2 MPa by a safety relief valve (SRV) simulator. The ADS is actuated at 162 s and a rapid depressurization begins. As a result, the LPF begins. The LPCS and LPCI initiate water injection at 330 s and 426 s, respectively.

The pressure transient of Run 934 is quite similar to the pressure transient of Run 922. The pressure recovery after the MSIV closure is not so large as in Run 922 because the core steaming rate is less due to less power input. However, once the ADS valve opens, the pressure transient for the two tests are the same. The ADS flow rate is so large compared to the core steaming rate that the difference between the power inputs of Run 922 and Run 934 has a minor effect on the pressure transient. The ADS flow rate at 180 s is 1.1 kg/s in both experiments. The vaporization rate by heat generation at 180 s is 0.25 kg/s in Run 922 and 0.17 kg/s in Run 934, respectively. Also, the difference between the two power curves becomes zero after 260 s since the initial stored energy is almost completely released by 260 s\textsuperscript{1,2,4}. The decay power becomes the only power source after 260 s in Run 922 and after 200 s in Run 934, respectively.

The mixture level behavior of both cases is compared in Fig. 10.11. In both cases uncovering occurs at the top part of the core before the LPF. Following the LPF, the core begins to be uncovered from top to bottom. After initiation of the LPCS, first the lower plenum and then the core begins to be filled with LPCS water. The core mixture level restoration is accelerated after the initiation of the LPCI. The downcomer mixture level decrease is earlier than the core mixture level decrease and the restoration is a little later than the core mixture level restoration. The downcomer mixture level in Run 934 recovers by the flashing in the downcomer, whereas it is not observed in Run 922. The level fall in the downcomer in Run 922 is faster than in Run 934 due to larger break flow by higher system pressure, therefore little liquid remains in the downcomer when the flashing begins in the downcomer. There is not any significant difference between the mixture level transients
of Run 922 and Run 934 in the pressure vessel.

The surface temperatures of the heater rod A-11 in Runs 922 and 934 are compared in Fig. 10.12. The trends of the core mixture levels are also shown. In Run 922 the surface temperature begins to rise at the top of the core at 144 s and the surface is rewetted at 169 s due to the LPF at 162 s. The surface temperature excursion propagates from the top to the bottom of the core with the core mixture level decrease. Water falls into the core after LPCS initiation and begins to cool the heater rods. Quenching propagates downward from the top of the core and also upward from the bottom of the core as LPCS water accumulates in the core. The bottom-up quenching becomes prominent after the initiation of the LPCI and the mid-core is finally quenched at 457 s. The surface temperature transient corresponds well to the core mixture level transient.

The general trends of the surface temperature transient in Run 934 are the same as in Run 922. Dryout and quenching in both cases occur at the same time at the same elevation. The slopes of the surface temperature increase and the maximum temperatures in both cases are also the same at the same elevation.

10.4 Discussion

One of the most important quantities in the safety assessment of a LWR is the PCT during a LOCA. The nuclear reactor safety licensing criteria stipulates that the PCT should not exceed 1473 K under any situation\(^8\).

The PCTs obtained in this test series are summarized in Table 10.5 and compared in Fig. 10.13. PCT results which were obtained by changing the break area in the range from 0 through 200% at the recirculation pump inlet side with the failure of the HPCS diesel generator\(^9\) (see Chapter 5) are incorporated into the figure. In Reference 9 the conservative power input curve (Base Curve) was used. The break area is normalized using the recirculation line flow area. The PCTs of the Realistic Curve case for both break areas of 200 and 50% are much lower than the PCTs of the Base Curve case. The PCT of the Realistic Curve case for a break of 5% agrees with the PCT of the Base Curve case for a break area of 5% with the difference of only 31 K.

In a large break LOCA such as a 200% break LOCA, the transient time is short. Core recovery occurs before the initial stored heat in a fuel rod has been fully released. Thus, the differences between the ANS Curve case; i.e. no initial stored heat case, the Realistic Curve case; i.e. realistic initial stored heat case, and Base Curve case; i.e. conservative initial stored heat case, are significant. The larger core power (larger stored heat) in the Base Curve case vaporizes more liquid in the core than in the Realistic Curve case. Thus, earlier and longer fuel rod recovery occurs in the Base Curve case than in the Realistic Curve case. After recovery, the higher power in the Base Curve case causes a larger heatup rate and a higher heater rod surface temperature than in the Realistic Curve case. The longer recovery in the Base Curve case also provides a longer heatup period. The difference between the PCTs of the Realistic and ANS Curve cases can be also explained in the same manner. Thus, the initial stored heat in a fuel rod is an important factor in a large break LOCA.

In a small break LOCA such as a 5% break LOCA, the transient is slow and long. Most part of the initial stored heat in a fuel rod is released before the core recovery occurs. After the stored energy is released, the decay power is the main source of the heat from a fuel rod to coolant. Thus, the initial stored heat in the fuel rod is small compared with the integrated heat generation in the core during the transient. It is concluded that the initial stored heat in a fuel rod has little effect on the surface temperature transient during a 5% break LOCA.
The dryout and quenching have inherently statistical characteristics. Thus, it is hard to say whether the PCT difference between the Base and Realistic Curve cases at 5% break tests which is shown in Fig. 10.13 has systematic reason or not.

In the Realistic Curve case, the PCT decreases with increase in the break area from 5% to 200%. The heater rod surface temperature increase is determined by the balance between the heat generation and the heat transfer to coolant. The deterioration of heat transfer to coolant is closely related to the core uncoverage. As the break size increases, the depressurization becomes faster and the LPCS and the LPCI are actuated earlier, which results in decreases in the core uncoverage duration and the PCT. It suggests that the most effective way for reducing the PCT is to initiate the injection of ECCS water as soon as possible.

In a small break LOCA less than 5%, the transient becomes slower than the transient of a 5% break LOCA. The initial stored heat (which is simulated by power input to fuel rods in the ROSA-III test) will be released before the core uncoverage and the effect of stored heat on the core uncoverage can be neglected. The only heat source which should be considered for predicting the PCT is the decay power. In this sense, the Base Curve case proves to be the Realistic Curve case. It is shown in Fig. 10.13 that the PCTs in the Base Curve tests decrease with the decrease in the break size below 5% because of the later uncoverage and the lower decay power, even through the duration of the uncoverage increases for the smaller break size. The result is expected in the Realistic Curve case.

In the Base Curve case the maximum PCT is observed at the 50% break test. In the small break test of 5%, the initial stored heat has little effect on the surface temperature rise. In the large break test of 200%, the stored heat has much effect on the surface temperature rise, however, the LPCS and the LPCI initiate water injection shortly after the initiation of temperature excursion and there is not enough time to heat up the fuel rod. In the 50% break test, the occurrence of the uncoverage is not delayed so much from that in the 200% break test that the stored heat is not fully released by the time. The transient is slow compared with the 200% break test and there is enough time to heat up the fuel rod. The larger stored heat in the Base Curve case than in the Realistic case provides higher heat-up rate. These effects mingle to result in the maximum PCT at the 50% break test.

It is concluded from the above discussions that the ROSA-III PCT results with the Base Curve correspond to the PCTs of the BWR LOCAs in break areas less than 5% and provide the upper bounds of the PCTs of the BWR LOCAs in break areas larger than 5%. The ROSA-III PCT results with the Realistic Curve also provide the lower bounds of the PCTs of the BWR LOCAs in the break areas larger than 5%.

10.5 Conclusions

The following conclusions have been obtained:

(1) In a large break LOCA with a break area larger than 50%, cladding surface temperature excursions are closely related with the initial stored heat in the fuel rod because core uncoverage occurs before the initial stored heat is released to coolant. In a 200% break LOCA experiment, the entire core was uncovered and the PCT reached 785 K when the Base Curve was used. When the Realistic Curve was used the duration of entire core uncoverage was shorter than the Base Curve case and the PCT was 617 K. When the ANS Curve was used, the bottom part of the core was covered with two phase fluid during the transient. The PCT was 581 K. In the 50% break LOCA experiments, the PCTs were 925 K and 785 K when Base Curve and Realistic Curve were used, respectively. Larger quantity of initial stored heat translates into the higher PCT in a large
break LOCA.

(2) In a small break LOCA with a break area smaller than 5%, the initial stored heat in fuel rods has little effect on the temperature excursion of the heater rod and the core mixture level transient because the core recovery occurs after the initial stored heat is released. The PCTs were 835 K and 866 K when the Base and Realistic Curves were used, respectively.

(3) When the Base Curve was used, the maximum PCT was obtained in the 50% break test. When the Realistic Curve was used, the maximum PCT was obtained in the 5% break test.

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Fig. 10.13 Peak cladding temperature spectrum in ROSA-III power curve sensitivity test series
### Table 10.1 Summary of test conditions

<table>
<thead>
<tr>
<th>Run Number</th>
<th>Run 926</th>
<th>Run 907</th>
<th>Run 906</th>
<th>Run 916</th>
<th>Run 933</th>
<th>Run 922</th>
<th>Run 934</th>
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<tr>
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<td>200%</td>
<td>50%</td>
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<td>5%</td>
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<td>realistic curve</td>
<td>ANS curve</td>
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<td>realistic curve</td>
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<td>realistic curve</td>
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<td>71 (s)</td>
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<td>LPCI</td>
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<td></td>
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<td>96 (s)</td>
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* Runs were conducted with the assumption of HPCS diesel generator failure.

### Table 10.2 Sequence of major events in 200% break tests

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<td>Steam line closure</td>
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<td>Recovery of jet pump suction</td>
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<td>Recovery of recirculation line outlet nozzle</td>
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<td>Initiation of lower plenum flashing</td>
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<td>Initiation of feedwater line flashing</td>
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<td>LPCS initiation</td>
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<tr>
<td>LPCI initiation</td>
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<tr>
<td>ADS actuation</td>
<td>131</td>
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<tr>
<td>Final quenching of core</td>
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### Table 10.3  Sequence of major events in 50% break tests

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<th>Events</th>
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<td>Uncovery of jet pump suction</td>
<td>13</td>
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<tr>
<td>Uncovery of recirculation line outlet nozzle</td>
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<tr>
<td>Initiation of lower plenum flashing</td>
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<tr>
<td>ADS actuation</td>
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<td>Initiation of feedwater line flashing</td>
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### Table 10.4  Sequence of major events in 5% break tests

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<td>Initiation of lower plenum flashing</td>
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<tr>
<td>Uncovery of recirculation line outlet nozzle</td>
<td>162</td>
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<tr>
<td>LPCS initiation</td>
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<td>Initiation of feedwater line flashing</td>
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### Table 10.5  Comparison of peak cladding temperatures

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<th>Run number</th>
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<th>Power curve</th>
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<th>Temperature (K)</th>
<th>Rod position</th>
<th>Rod number</th>
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<tr>
<td>926</td>
<td>200</td>
<td>Base Curve</td>
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<td>Realistic Curve</td>
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<td>617</td>
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<td>Realistic Curve</td>
<td>411</td>
<td>866</td>
<td>Pos. 4</td>
<td>A82</td>
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</table>
Fig. 10.1 Comparison of BWR and ROSA-III stored heat term in heat transfer rate from fuel rod surface to coolant.

Fig. 10.2 Comparison of Base, Realistic and ANS Curves.
Fig. 10.3 System pressures in 200% break experiments.

Fig. 10.4 Vessel side break flows in 200% break experiments.
Fig. 10.5  Mixture levels in pressure vessel in 200% break experiments.
Fig. 10.6 Fuel rod surface temperatures in 200% break experiments.
Fig. 10.7  System pressures in 50% break experiments.

Fig. 10.8  Mixture levels in pressure vessel in 50% break experiments.
Fig. 10.9 Fuel rod surface temperatures in 50% break experiments.
Fig. 10.10  System pressures in 5% break experiments.

Fig. 10.11  Mixture levels in pressure vessel in 5% break experiments.
Fig. 10.12 Fuel rod surface temperatures in 5% break experiments.
Fig. 10.13  Peak cladding temperature spectrum in ROSA-III power curve sensitivity test series.
11. Effect of Differences in Heat Generation among Fuel Bundles

11.1 Introduction

Four LOCA tests were performed at the ROSA-III facility in order to examine the interaction of thermal-hydraulic phenomena between bundles in a multi-bundle core. The experiments consist of two large break LOCA tests and two small break LOCA tests with/without heat generation difference among the four bundles.

11.2 Test Description

The test conditions of the four tests which are discussed in this chapter are summarized in Table 11.1. In each test the failure of the HPCS diesel generator was assumed and no ECCS water was injected from the HPCS. The break location was at the recirculation pump inlet line and the break type was 200% double-ended break in Runs 926 and 981 and 5% split break in Runs 922 and 982. The scaled recirculation line flow area from the BWR/6 corresponds to 100% break size, Blowdown was initiated by opening the quick opening blowdown valve(s).

In Runs 926 and 922, the power supplied to Bundle A was 1.4 times larger than that to Bundles B, C or D. In contrast, the power supplied to Bundle A was the same as that to Bundle B, C or D in Runs 981 and 982.

The primary test initial conditions before break were as follows. The steam dome pressure was 7.35 MPa and the corresponding saturation temperature was 562 K. The core inlet flow rate was 16 kg/s and the core outlet quality was 14%. The lower plenum subcooling was 11 K.

Liquid level signals in the downcomer were used to initiate main steam isolation valve (MSIV) closure and ECCS actuation in the tests. The initial liquid level in the downcomer was 5.00 m since the liquid volume in the downcomer below the elevation of 5.00 m in ROSA-III, including the volume in the jet pump suction pipings, corresponded to the volume below the scram level (L3 level) in a BWR/6. The L2 and L1 liquid levels in the downcomer of ROSA-III were 4.76 m and 4.25 m, respectively. The MSIV closure was initiated by the L2 level signal with a time delay of 3 s. The LPCS, LPCI and ADS actuation was initiated by the L1 level signal after 40 s, 40 s and 120 s time delay, respectively. The same delay is used in the safety analysis of a BWR. The LPCS and LPCI injections were further specified to be initiated at system pressures of 2.2 MPa and 1.6 MPa, respectively.

The safety relief valve (SRV) was operated in the 5% break experiments so the vessel pressure might not exceed 8.14 MPa.

11.3 Test Results and Discussions

11.3.1 200% Double-Ended Break

The vessel pressure transients in Runs 926 and 981 are shown in Fig. 11.1. The difference between the two pressure transients is very slight. The principal trend of Run 926 is summarized in Table 11.2. The vessel pressure in each test decreases immediately after the break initiation as fluid is discharged. The pressure temporarily increases after the closure of the MSIV but again starts to decrease due to the discharge of fluid and the decay of heat genera-
tion in the core. The LPCS and the LPCI begin fluid injection into the vessel when the vessel pressure reaches 2.2 MPa and 1.6 MPa, respectively.

Mixture levels in Bundles A, B, C and D and the lower plenum for Runs 926 and 981 are shown in Fig. 11.2. The mixture levels were measured by conduction probes. The core mixture level falls in both experiments because of mass loss from the break hole and the steam line. As a result, the core finally dries out. After the coolant injection initiation from the LPCI, the core mixture level recovers. The mixture level in the lower plenum falls due to vaporization of fluid caused by the vessel depressurization but recovers in both tests after the LPCS coolant injection initiation.

The CCFL phenomenon is observed at the upper tieplate, the lower tieplate and the channel inlet orifice for certain periods. The CCFL phenomenon is characterized by liquid retention by flowing-up vapor above the place where the flow path is contracted, even if the downside space of the contracted flow path is filled with vapor.

The mixture level transitions in both experiments are quite similar except that a temporary mixture level fall from 10 to 20 s is observed at the upper part of the core in Run 926. This is not observed in Run 981, which suggests that the four bundles closely interact with each other. It should be also noted that the temporary mixture level fall is observed in all bundles in Run 926; even in Bundles B, C and D which have the same heat generation rate as those in Run 981. It should also be noted that the difference of the core mixture level behavior among the bundles in Run 981 is smaller than that in Run 926 although the core mixture level fall and recovery is quite similar in both tests.

Figure 11.3 shows the differential pressures measured at the channel inlet orifices in Runs 926 and 981. The flow at the orifices is two-phase during almost the whole transient period except during the initial 17 s. Thus, the value itself has little meaning in connection with translating it into flow rate. However, the flow direction at the orifices can be identified. It is interesting that the flow directions at the channel inlet orifices of Bundles A, B, C and D are the same until the LPCI injection initiation and after that some channels show upward flow and others show downward flow. This phenomena is observed in both tests, however, there is no fixed trend as to which channel has upward or downward flow.

Figure 11.4 shows cladding surface temperatures measured in Runs 926 and 981. Rods A-11 and C-11 shown in the figure are located at the corner of Bundles A and C, respectively. The rods are instrumented with thermocouples at seven elevations.

In Run 926, a temporary temperature excursion is observed at the upper part of the core in the early stage of the transient. The temperature excursion is quenched by the core mixture level swell which is caused by lower plenum flashing. The temporary temperature excursion is not clearly observed in Run 981. This phenomenon corresponds to the core mixture transient discussed for Fig. 11.2.

The difference of the main temperature excursions between Run 926 and Run 981 is quite clear. In Run 981, the heat generation is the same in all the bundles, the cladding temperatures of Rods A-11 and C-11 behave quite similarly and dryout and quenching occur at nearly the same respective times in both channels. In contrast, in Run 926 the dryout of Rods A-11 and C-11 occurs at different times except in the lower part of the core. The quenching times are quite different between Rods A-11 and C-11 in Run 926. The difference in the rate of temperature increase after dryout is attributed to the difference of the heat generation rate between the rods.

11.3.2 5% Break

The vessel pressure transients in Runs 922 and 982 are quite similar as shown in Fig. 11.5.
In both, the vessel pressure begins to decrease because of fluid discharge from the break and heat generation decay in the rods after the initiation of break. After the MSIV closure the pressure increases until the SRV set pressure is reached. In both runs, the pressure is kept nearly constant by the SRV open-close cycling until the operation of the ADS. After the ADS operation, the vessel pressure decreases rapidly and the LPCS and the LPCI are initiated to inject coolant into the vessel at the vessel pressures of 2.2 MPa and 1.6 MPa, respectively.

Mixture levels in the core and the upper and lower plena, measured by conduction probes, for Runs 922 and 982 are shown in Fig. 11.6. The mixture level transients in both experiments are almost the same. The core mixture levels in both experiments begin to fall because of mass loss from the break hole and the steam line. The core is temporarily filled up again with the two-phase mixture due to mixture level swell caused by the rapid depressurization after the ADS actuation. As the transient progresses, the level again falls and the core finally dries out. After the initiation of the LPCS and the LPCI, the core is filled again with the two-phase mixture. The core mixture level recovery in Run 922 is a little earlier than in Run 982 because the LPCS and the LPCI began to inject coolant a little earlier in Run 922. The total heat generation rate in the core in Run 982 was lower than in Run 922, thus the depressurization rate in Run 982 was slightly faster than in Run 922, which resulted in slightly earlier initiation of the LPCS and LPCI coolant injections.

The mixture levels in both experiments fall continuously from the upper plenum to the lower plenum and recover in the reverse order. Unlike the 200% break experiments, no CCFL phenomenon was observed. It should be noted that the mixture levels in the four bundles behave almost the same in both tests.

Differential pressures measured at the channel inlet orifices in Runs 922 and 982 are shown in Fig. 11.7. In both tests, liquid drainage from the Channel A is observed from 43 to 173 s, whereas the flow in Channels B, C and D is upward or nearly stagnant during this period. The flow rate of the upward flow or downward flow may be very small because all the measured differential pressures are small. However, after the initiation of the LPCI injection, flow is clearly upward in some channels and downward in other channels. There is no preference as to which bundles have upward or downward flow.

Cladding surface temperatures measured in Runs 922 and 982 are shown in Fig. 11.8. Data shown in the figure were measured on Rods A-11 and C-11.

As expected from the mixture level transient, the cladding temperature transients of Rod A-11 and Rod C-11 are quite similar until the temperature excursion in Run 922 or 982. In Run 922, after the temperature excursion occurs, the rate of temperature increase of Rod A-11 is larger than that of C-11 because the heat generation in Bundle A was 1.4 times larger than that in Bundle C. In Run 982, there was no difference between the heat generation in Bundles A and C, and the rates of temperature increase of Rods A-11 and C-11 after dryout are nearly the same. It is interesting to note that the quenching times of Rods A-11 and C-11 are quite different in Run 922, whereas they are nearly the same in Run 982. This will be discussed further in the following section.

11.3.3 Discussions

Core thermal-hydraulics in Runs 926, 981, 922 and 982 will be discussed in this section. The discussion will focus on the dryout or quenching propagation difference among rods in a bundle or among bundles in the core.

Dryout and rewetting/quenching propagation traces measured in Runs 926, 981, 922 and 982 are shown in Figs. 11.9 through 11.12.

The temporary dryout and rewetting, for example, from 4 to 26 s and from 65 to 100 s,
respectively, in Bundle A in Run 926 as shown in Fig. 11.9(a), are observed in all the experiments although the degree of extent is different. These dryout and rewetting phenomena may have some effect on the PCT during the LOCA transient. However, the controlling (and interesting) phenomena relating to the PCT are the phenomena related to main dryout and quenching, for example, from 36 to 72 s and from 100 to 184 s, respectively, in Bundle A in Run 926 as shown in Fig. 11.9(a). Therefore, only the main dryout and quenching phenomena are discussed in the following.

From Fig. 11.9(a), it is noted that the difference in the propagation of the main dryout is very small among rods in Bundle A in the 200% break test, Run 926, in which a heat generation difference existed between Bundle A and Bundles B, C or D. However, the propagation of quenching shows a large scattering from rod to rod in the figure. Quenching time at the same elevation is widely scattered from rod to rod. Some rods are quickly quenched from the top or bottom. Quenching of other rods propagates slowly. This tendency is the same in Bundles B, C and D in Run 926, as shown in Fig. 11.9(b).

The dryout propagation difference between Bundle A and Bundles B, C or D in Run 926 is little and the dryout occurs at the same time period for Bundles A, B, C and D. The quenching propagation in the Bundles A, B, C and D is quite similar in Run 926. The quenching seems to occur a little earlier in Bundles B, C and D than that in Bundle A.

Run 981 was a 200% break test with no heat generation difference between Bundles A, B, C and D. The same statements as in Run 926 are valid as to the dryout and quenching propagation, as shown in Fig. 11.10. Only the difference which can be observed is that the quenching is completed in a shorter period in Run 981 than in Run 926.

The same tendency as in the 200% break tests of Runs 926 and 981 is observed in the 5% break experiments of Runs 922 and 982, as shown in Figs. 11.11 and 11.12.

Tables 11.3 and 11.4 clarify the above statements. Differential time, \((T_q)_r - (T_q)_q\), between the last quenching and the first quenching at each elevation in Bundle A and Bundles B, C and D, the average of the quenching time and the standard deviation of the quenching time are included in the tables.

Table 11.3 is for the 200% break tests of Runs 926 and 981. The average of the quenching time for Bundle A is a little later than that for Bundles B, C and D in Run 926 in which the heat generation difference existed between Bundle A and Bundles B, C and D. In Run 981, in which there was no heat generation difference among the bundles, the averages of the quenching time for Bundle A and Bundles B, C and D are nearly the same. It is also noted that the standard deviation of the quenching time for Bundle A is larger than that for Bundles B, C and D in Run 926, which implies that the quenching time scattered more widely from rod to rod in Bundle A than in Bundles B, C and D. In Run 981, the standard deviation for Bundle A and Bundles B, C and D are nearly the same, which implies that the scattering of quenching time is nearly the same in Bundle A and Bundles B, C and D. It is also interesting to note that the standard deviation for Bundle A in Run 981 is smaller than that in Run 926. This tendency is also the same as for Bundles B, C and D. It suggests that the scattering of the quenching propagation in a bundle decreases if the heat generation difference among bundles decreases or disappears.

Table 11.4 is for the 5% break tests of Runs 922 and 982. It should be kept in mind that even if the heat generation difference exists between Bundle A and Bundles B, C and D, time interval from break initiation to dryout occurrence is much longer than that in the 200% break experiments. Thus, the absolute value of the heat generation difference in the dryout period is smaller in the 5% break tests than that in the 200% break tests.

In Run 922 in which the heat generation difference existed between Bundle A and
Bundles B, C and D, the average of the quenching time for Bundle A is a little later than that for Bundles B, C and D. In Run 982, in which there was no heat generation difference among the bundles, the averages of the quenching time for Bundle A and Bundles B, C and D are nearly the same. These results are the same as the 200% break experiments.

The standard deviations of the quenching time for Bundle A and Bundles B, C and D in Run 922 show that the scattering of the quenching time is much larger in Bundles B, C and D than that in Bundle A. This result is opposite of the result of the 200% break experiment. However, it is the same result as for the 200% break experiment that the scattering of the quenching time becomes small in Bundle A and Bundles B, C and D and the scattering becomes nearly the same in all bundles when the heat generation difference among the bundles disappears.

It can be concluded from the above observations that the axial propagation of quenching becomes uniform in the same bundle as well as in the core in the large and small break tests when the heat generation difference among the bundles disappears. The results indicate that even small heat generation differences among the bundles create flow imbalance among the bundles because of void fraction difference which is caused by the different heat generation rates and the different vapor generation rates among bundles. Although the flow imbalance is even less prominent as in the present experiment series, as shown in Figs. 11.3 and 11.7, the slight flow imbalance may have much effect on the quenching propagation.

### 11.4 Conclusions

Two 200% large break and two 5% small break loss-of-coolant tests with and without heat generation difference among the bundles were conducted at the ROSA-III test facility to investigate the interaction of thermal-hydraulic phenomena among bundles in a multibundle core. The following conclusions were obtained from the investigation of the test results:

1. Flow imbalance among the bundles was not prominent during the blowdown and reflooding periods both in the large and small break experiments regardless of the heat generation difference among bundles. After the reflooding was completed, some bundles showed upward flow and other bundles showed downward flow in each experiment. However, there was no preference of flow direction in each bundle.

2. The radial distribution of the axial propagation of quenching tends to become more pronounced in the same bundle as well as in the core in both large and small break experiments when the heat generation difference exists among the bundles. Although the flow imbalance was not clearly observed in the experiments, even the slight imbalance in flow among the bundles, which is caused by void fraction difference due to the heat generation difference among the bundles, is considered to increase the scattering of the quenching propagation.

### References

11. Effect of Differences in Heat Generation among Fuel Bundles

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(b) Bundles B, C and D
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(a) Bundle A
(b) Bundles B, C and D
Fig. 11.12 Dryout and quenching propagation traces in 5% break test, Run 982
(a) Bundle A
(b) Bundles B, C and D
Table 11.1 Test conditions

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Run 926</th>
<th>Run 981</th>
<th>Run 922</th>
<th>Run 982</th>
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<tr>
<td>Break Area</td>
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<td>1.4</td>
<td>1.0</td>
<td>1.4</td>
</tr>
<tr>
<td>Power Distribution in Core</td>
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<td></td>
<td></td>
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<td>A : 1.26 MW</td>
<td>A, B, C, D : 0.9 MW</td>
<td>A : 1.26 MW</td>
<td>A, B, C, D : 0.9 MW</td>
<td>A, B, C, D : 0.9 MW</td>
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<tr>
<td>B, C, D : 0.9 MW</td>
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<td></td>
<td></td>
</tr>
</tbody>
</table>

* HPCS : Not actuated.
* LPCS : Initiated at system pressure of 2.2 MPa with L1 signal.
* LPC1 : Initiated at system pressure of 1.6 MPa with L1 signal.
* ADS : Actuated with L1 signal with time decay of 120 s.
* MSIV closure : Closed with L2 signal with time delay of 3 s.
* SRV : Opened at system pressure above 8.14 MPa.

Table 11.2 Chronology of events

(a) Run 926 (200% break)

<table>
<thead>
<tr>
<th>Time after break s</th>
<th>Events</th>
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<tbody>
<tr>
<td>0</td>
<td>Break</td>
</tr>
<tr>
<td>3</td>
<td>MRPs started coast-down</td>
</tr>
<tr>
<td>4</td>
<td>L2 level signal</td>
</tr>
<tr>
<td>8</td>
<td>Feedwater stopped</td>
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<td>9</td>
<td>L1 level signal</td>
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<td>9</td>
<td>Initiation of core power decay</td>
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<tr>
<td>21</td>
<td>Main steam line closed</td>
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<tr>
<td>71</td>
<td>LPCS actuation</td>
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<tr>
<td>96</td>
<td>LPC1 actuation</td>
</tr>
<tr>
<td>131</td>
<td>ADS opened</td>
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<tr>
<td>688</td>
<td>Experiment terminated</td>
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(b) Run 982 (5% break)

<table>
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<th>Time after break s</th>
<th>Events</th>
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<td>MRPs started coast-down</td>
</tr>
<tr>
<td>9</td>
<td>Feedwater stopped</td>
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<tr>
<td>21</td>
<td>L2 level signal</td>
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<td>25</td>
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<td>42</td>
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<tr>
<td>75</td>
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<tr>
<td>162</td>
<td>ADS valve opens</td>
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<td>Initiation of lower plenum flashing (LPF)</td>
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<tr>
<td>426</td>
<td>LPC1 actuation</td>
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<tr>
<td>959</td>
<td>Experiment terminated</td>
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Table 11.3 Quenching time in 200% break tests

<table>
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<th>Bundle</th>
<th>Run 926</th>
<th>Run 981</th>
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<tr>
<td></td>
<td>$(T_q)_L - (T_q)_R$</td>
<td>$T_q$</td>
</tr>
<tr>
<td></td>
<td>(s)</td>
<td>(s)</td>
</tr>
<tr>
<td>Position 1</td>
<td>A B,C,D</td>
<td>64 41</td>
</tr>
<tr>
<td>Position 2</td>
<td>68 35</td>
<td>149.1 135.3</td>
</tr>
<tr>
<td>Position 3</td>
<td>72 53</td>
<td>148.8 131.9</td>
</tr>
<tr>
<td>Position 4</td>
<td>19 22</td>
<td>132.1 122.1</td>
</tr>
<tr>
<td>Position 5</td>
<td>16 10</td>
<td>117.4 113.0</td>
</tr>
<tr>
<td>Position 6</td>
<td>6 5</td>
<td>108.6 107.5</td>
</tr>
<tr>
<td>Position 7</td>
<td>23 20</td>
<td>96.0 100.3</td>
</tr>
</tbody>
</table>

$T_q$ : Average quenching time
$(T_q)_R$ : First quenching time
$(T_q)_L$ : Last quenching time
$\sigma$ : Standard deviation of quenching time

Table 11.4 Quenching time in 5% break tests

<table>
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<th>Bundle</th>
<th>Run 922</th>
<th>Run 982</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>$(T_q)_L - (T_q)_R$</td>
<td>$T_q$</td>
</tr>
<tr>
<td></td>
<td>(s)</td>
<td>(s)</td>
</tr>
<tr>
<td>Position 1</td>
<td>A B,C,D</td>
<td>108 220</td>
</tr>
<tr>
<td>Position 2</td>
<td>105 125</td>
<td>366.1 365.7</td>
</tr>
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<td>Position 3</td>
<td>98 115</td>
<td>385.6 371.6</td>
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<tr>
<td>Position 4</td>
<td>87 105</td>
<td>408.0 383.7</td>
</tr>
<tr>
<td>Position 5</td>
<td>91 97</td>
<td>420.8 390.5</td>
</tr>
<tr>
<td>Position 6</td>
<td>87 92</td>
<td>419.9 392.2</td>
</tr>
<tr>
<td>Position 7</td>
<td>33 36</td>
<td>353.8 352.6</td>
</tr>
</tbody>
</table>

$T_q$ : Average quenching time
$(T_q)_R$ : First quenching time
$(T_q)_L$ : Last quenching time
$\sigma$ : Standard deviation of quenching time
Fig. 11.1 Vessel pressures in 200% break tests Runs 926 and 981.

(a) With heat generation difference among bundles, Run 926

(b) Without heat generation difference among bundles, Run 981

Fig. 11.2 Mixture levels inside shroud in 200% break tests.
Fig. 11.3  Differential pressures at channel inlet orifices in 200% break tests.

Fig. 11.4  Heater rod surface temperatures in 200% break tests.
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Fig. 11.6  Mixture levels inside shroud in 5% break tests.
Fig. 11.7  Differential pressures at channel inlet orifices in 5% break tests.

Fig. 11.8  Heater rod surface temperatures in 5% break tests.
Fig. 11.9  Dryout and quenching propagation traces in 200% break test, Run 926.

(a) Bundle A

(b) Bundles B, C and D

Fig. 11.10  Dryout and quenching propagation traces in 200% break test, Run 981.
Fig. 11.11 Dryout and quenching propagation traces in 5% break test, Run 922.

Fig. 11.12 Dryout and quenching propagation traces in 5% break test, Run 982.
12. Break Configuration Parameter Tests

12.1 Introduction

The effects of break configuration\(^1\) on LOCA transients are discussed in this chapter. The break configuration is important because the break flow rate depends not only on the break flow area but also on the break configuration, and the effects of the break flow rate on LOCA transients are significant. Fauske\(^2\) indicated that subcooled water critical flow rate decreases with increase in ratio of throat length to throat diameter (L/D ratio). Henry\(^3\) discussed that this tendency was significant in the subcooled water critical flow, however, the configuration difference had little effect on two-phase or single phase vapor critical flow rate.

It is hard to identify what break configuration would appear in a BWR LOCA. In order to apply the experimental results of the ROSA-III, which were obtained by using a specified break configuration, to a BWR LOCA in which any break configuration may appear, it is necessary to investigate and clarify the effects of the break configuration on the system transient during a LOCA. The break configuration sensitivity series of experiment were conducted at the ROSA-III test facility by using a sharp-edged orifice and a long throat nozzle at the break. Six LOCA experiments with break flow area of 15%, 50% or 200% were conducted.

The results of the tests and the RELAP4/MOD6/U4/J3 analyses are presented in this chapter. Details are given in Ref. 1.

12.2 Test Conditions

The test conditions are summarized in Table 12.1. The break location was fixed at the main recirculation pump inlet piping in the test series. The break size was 200% in Runs 964 and 904, 50% in Runs 916 and 928 and 15% in Runs 913 and 927. A quick shut-off valve was closed immediately after the break to simulate a double-ended break in Runs 964 and 904. A single-ended break was simulated in the other tests without closing the valve. Two kinds of break configuration were used, a sharp-edged orifice and a long throat nozzle as shown in Fig. 12.1. The orifice was used in Runs 904, 916 and 913 and the nozzle was used in Runs 964, 928 and 927. HPCS diesel generator failure was assumed in the test series.

12.3 Test Results

12.3.1 Chronology of Major Events
Chronology of major events in this test series is listed in Table 12.2.

12.3.2 Break Flow
Figure 12.2 shows the test results of break flow mass fluxes. The break flow rates were measured by the combination of drag disks and gamma densitometers. The mass fluxes of 200% loss of coolant experiments are the pressure vessel side break flow mass fluxes. Each mass flux transient shows similar trend. The mass flux was nearly constant after the break until recirculation line uncoverey (RLU). During this period, fluid upstream of the break was
single phase liquid. The mass flux rapidly decreased immediately after the fluid upstream of the break turned to two-phase or single-phase vapor after RLU. Figure 12.1 shows clear differences between the orifice mass flux and the nozzle mass flux in 15% and 50% tests, respectively. The orifice mass fluxes were approximately 1.5 times larger than the nozzle mass fluxes. This result agrees with the data in Refs. 2 and 3. The reason is supposed to be thermodynamic nonequilibrium effect. The pressure decreases as the fluid approaches to the throat because the fluid velocity increases due to the flow area contraction. When the evaporation or bubble nucleation does not initiate or increase enough to maintain thermal equilibrium after the fluid pressure falls below the saturation pressure, a thermally nonequilibrium flow is obtained at the throat. Nonequilibrium flow rate is larger than the equilibrium flow rate because of higher fluid density. This effect is dependent on the time required for recovery of the equilibrium state and the elapsed time to go through the throat region. The former becomes longer with the increase in subcooling, and the latter becomes shorter with the decrease in throat length. Thus the nonequilibrium effect was more significant in the subcooled break flow through the orifice.

Figure 12.3 shows the measured discharge mass fluxes before RLU through the orifices vs. pressures measured at the 0.19 m upstream side of the break. Mass fluxes calculated by an orifice equation,

\[ W = C_0 \sqrt{2 \Delta P \rho} \]

with a flow coefficient of 0.61 are also shown in the figure. Here \( W \), \( C_0 \), \( \rho \), and \( \Delta P \) are the mass flux, flow coefficient, liquid density and pressure difference across the orifice, respectively. Pressures measured upstream and downstream of the break were used in the calculation. The data for initial 10 s in Run 913 are not plotted in the figure, because the mass flux transient in that period was peculiar as shown in Figure 12.2. The reason may be the instability of the drag disk which was caused by the rapid change in the break flow rate. The measured subcooling of fluid upstream of the break ranged between 3 and 14 K in Run 913, between 3 and 8 K in Run 916 and was nearly zero in Run 904.

In Run 916 of 50% test the measured mass fluxes agreed with the orifice equation as indicated in the literature. The mass fluxes measured in Run 904 of a 200% break test were less than the calculated results. This is because the fluid upstream of the break became saturated due to rapid depressurization. Saturated break flow rate through the orifice is considered smaller than subcooled flow as already described. In Run 913 of a 15% break test the measured results were larger than the orifice equation with \( C_0 \) of 0.61. The measured results correspond to the orifice equation with \( C_0 \) of 0.8. The disagreement of \( C_0 \) is considered to be caused by geometrical difference between the orifices used in Runs 913 and 916 as shown in Figure 12.1. The orifice used in Run 913 (ID 10.1 mm) has a flow area reduction region (ID 30.0 mm, length 32 mm) just upstream of the throat. This area reduction probably caused the smaller effect of flow contraction at the throat than in Run 916.

Figure 12.4 shows the measured subcooled discharge mass fluxes through the nozzles vs. subcooling, and calculated results by the Henry-Fauske model (HFM) and the homogeneous equilibrium model (HEM). Results measured by Lee et al.\(^4\) are also plotted in the figure. In their experiments nozzle diameter was 1.8 and 2.4 mm and the length-to-diameter (L/D) ratio was 3.57 and 3.48, respectively. The experimental pressure was 7 MPa. The data for the initial 10 s in Run 927 of the 15% break test are eliminated in the figure for the same reason given for the elimination of the data in Run 913. Figure 12.4 clearly shows the dependence of the break mass fluxes on the subcooling. The experimental data were between HFM and HEM. HEM gave about 20% smaller break flow mass fluxes and HFM gave about
20% larger than the measured mass fluxes through the nozzle. Lee's data were about 15% higher than the ROSA-III data. It was considered to be due to the difference in the $L/D$ ratios between the nozzles used in both tests.

The dependence of two-phase or steam break flow rate on the break configuration will be discussed in the following section because the measurement by drag disks and gamma densitometers in the two-phase or single-phase vapor condition does not have enough accuracy to make comparison of the orifice and the nozzle break flow rates possible.

### 12.3.3 Pressure Transients

Figure 12.5 shows the system pressure transients vs. time in the break configuration parameter test series. Initiation times of MSIV closure, RLU and ADS actuation, which were governed by the downcomer mixture level, were earlier in tests with the orifice than in tests with the nozzle because the downcomer mixture level decreased earlier in tests with the orifice. Depressurization rates after RLU, however, were not affected significantly by the break configuration difference. Thus, the difference in the whole transient time was not significant between tests with the orifice and with the nozzle. Depressurization rates after ADS actuation in the 15% break tests were not affected by break configuration difference because ADS flow area corresponds to the 35% area of the scaled main recirculation pump (MRP) inlet piping cross sectional area and much larger than the break area. Experimental results that the break configuration difference had little effect on the depressurization rates after RLU indicate that two-phase or vapor single phase break flow was not affected significantly by the break configuration difference.

### 12.3.4 Core Mixture Level

Figure 12.6 shows the mixture level transients vs. time in the high power channel in this test series. The mixture level transients were measured by conductivity probes. The timings of the level fall initiation, completion of core uncoverage and restoration of the core mixture level were earlier in tests with the orifice than those with the nozzle because of subcooled discharge flow difference and the resultant LPCS and LPC1 actuation timing differences. Figure 12.6 also shows that the event timing differences between the orifice and the nozzle case become greater with decrease in the break area. It is because the duration of the subcooled discharge flow, which was affected by the break configuration difference, became longer with the decrease in the break size. The timing difference in the whole core recovery between the two 15% break tests was 29 s, for example. However, the differences of the timings of major events between tests with the orifice and with the nozzle were very small compared with the whole transient time. There was not a significant difference in the surface dryout duration between tests with the orifice and with the nozzle in each break size.

### 12.3.5 Cladding Surface Temperature

Figure 12.7 shows the measured PCTs in the tests series. The PCT in the 200% tests with the orifice was 14 K higher than that with the nozzle. The PCT in the 50% break tests with the orifice was 29 K higher than that with the nozzle. In the 15% tests, PCTs were almost the same between the two tests. If the break sizes are the same, the surface dryout durations in the two tests with the orifice and the nozzle are almost the same as discussed in Section 3.4. Thus the difference in break configuration between the orifice and the nozzle did not create significant difference in PCTs.

The break configuration difference does not have significant effects on the system total behavior such as system pressure, core mixture level and cladding surface temperature if the
throat area is the same. The reasons for this are as follows; (1) The difference in break configuration affected only the subcooled break flow rate and the duration of the subcooled break flow was small compared with the whole transient time, (2) Depressurization rates after RLU were not significantly affected by the break configuration probably because two-phase or vapor single phase discharge flow rates were not affected significantly by the break configuration difference between the orifice and the nozzle, (3) ADS flow area corresponded to the 35% MRP inlet piping area and was large enough to mask the break area difference effects after ADS actuation for small break tests.

12.4 Analysis

By analyzing the results of the test series and BWR counterpart LOCAs using a LOCA analysis code, applicability of these test results to the BWR LOCAs was investigated. Main interest in the analyses was to confirm how the liquid phase break flow difference caused by the break configuration difference would affect the total system behavior during a hypothetical BWR LOCA. First, the post test analyses of the ROSA-III experiments, Run 913 and 927, which were 15% tests with the orifice and the nozzle, respectively, were performed to assess the code capability. Then the BWR LOCA transients were analyzed using the same calculation models.

The RELAP4/MOD6/U4/J3 was used in the analysis, which was an improved version of RELAP4/MOD6 by Japan Atomic Energy Research Institute (JAERI). The 15% tests, Runs 913 and 927, were chosen for the analyses because the effects of the break configuration difference on system transients were most prominent in this test series. Two 15% break BWR LOCA calculations were also conducted by assuming an orifice break configuration and a nozzle break configuration, respectively.

The break flow rate was calculated by a combination of the Henry Fauske model (HFM) and a homogeneous equilibrium model (HEM). The single phase liquid critical flow rate was calculated by HFM and the two phase or steam critical flow rate was calculated by HEM. A flow coefficient of 1.0 for the nozzle and 1.5 for the orifice was used in HFM. A flow coefficient of 1.0 for both the nozzle and the orifice was used in HEM. These combinations of the flow coefficients provided good agreement between the measured break flow rate and pressure, and calculated results by RELAP4/MOD6/U4/J3. These flow coefficients are not consistent with the discussion in section 3.2 because the calculated subcooling upstream of the break was smaller than the experimental data. RELAP4/MOD6/U4/J3 code does not have a capability to calculate the subcooled discharge flow rate by using the orifice equation. The discharge flow calculation model and the discharge flow coefficients used in the BWR/6 analyses were the same as those used in the ROSA-III analyses. Detailed information about the calculation models such as nodalization, selected optional models, boundary conditions and so on can be found in Ref. 6.

12.4.1 ROSA-III Test Analysis

The calculated pressures are compared in Figure 12.8 with measured ones in the 15% tests (Runs 913 and 927). The calculated pressure transients agreed well with the experimental data, although small discrepancies were observed. The effect of the liquid phase break flow difference between the orifice case and the nozzle case on pressure transient was calculated qualitatively well. Timing of RLU was earlier in the orifice case than in the nozzle case, which is consistent with the experiments.

Figure 12.9 shows the calculated cladding surface temperatures compared with the
measured data. The calculated maximum temperatures in Run 913 and 927 analyses are 131 K and 123 K higher than the measured results, respectively. This is because the heat transfer coefficient used in the calculation after LPCS actuation until quench is smaller than the experimental data. The effect of the difference in the single phase liquid break flow rate between the orifice and the nozzle break configurations was well reflected on the timing of dryout and quenching in the analysis as in the pressure calculations.

12.4.2 BWR LOCA Analysis

Figure 12.10 shows comparisons of the calculated BWR pressure transients with those calculated for ROSA-III tests. Pressures in the orifice case of the BWR analyses were kept constant at 7.5 MPa from 12 s until 58 s because the pressure setpoint for the safety relief valve actuation was lower than that used in the experiments. This resulted in earlier actuations of LPCS and LPC1 in the BWR analyses than in the ROSA-III analyses. However, the fundamental phenomena such as the depressurization immediately after the break, the pressure increase caused by the MSIV closure and the depressurization caused by the RLU and the ADS actuation in the BWR analyses were quite similar to the analytical results for the ROSA-III. The difference in the liquid phase break flow rates between the two BWR analyses had little effect on the pressure transients as in the ROSA-III analyses.

Figure 12.11 shows the comparison of the cladding surface temperatures calculated at the midplane of the core. The PCT calculated for the orifice case in the BWR analyses was 839 K, 99 K lower than that for the orifice case in the ROSA-III analyses. The difference in the timing of quenching in the BWR/6 analyses and the ROSA-III analyses was caused by the difference in the ECCs actuation timing. Major event timing of temperature transients in the orifice case were earlier than in the nozzle case in the BWR analyses as in the ROSA-III analyses.

The analysis results did not show any significant effects of liquid phase break flow differences on both the BWR and the ROSA-III transients. This was because the timing difference due to the liquid phase break flow difference were substantially small compared with the whole LOCA transient duration in both analyses.

12.5 Conclusions

The following conclusions were obtained from the experiments and analyses.

1) Measured subcooled break flow through the orifice was larger than that through the nozzle, however, two-phase or vapor single phase break flow was almost the same for both the orifice and the nozzle.

2) Measured subcooled break flow through the sharp-edged orifice agreed well with the results calculated using the orifice equation with a flow coefficient of 0.61. This is consistent with the results in the existing literatures.

3) The homogeneous equilibrium model and the Henry-Fauske model gave 20% smaller and 20% larger subcooled break flow rates than the measured flow rates through the nozzle, respectively.

4) The effects of the difference in the subcooled discharge break flow rate on the system behavior were not significant in the ROSA-III tests. The timing difference of break recovery between the experiments with the orifice and with the nozzle was small compared with the duration of the whole LOCA transient.

5) RELAP4/MOD6/U4/J3 code calculated the basic trend of the 15% loss of coolant experiments satisfactorily. The effect of a break configuration difference was calculated
qualitatively well.

(6) Results of the BWR 15% break LOCA analyses showed that the difference in liquid phase break flow rate between orifice and nozzle had also little effect on the system behaviors in a BWR as in the ROSA-III experiments and analyses.

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Fig. 12.11 Comparison of calculated cladding surface temperatures at core midplane in the two BWR cases and the two ROSA-III cases
### Table 12.1  Experimental conditions

<table>
<thead>
<tr>
<th>Break condition</th>
<th>Run 904</th>
<th>Run 964</th>
<th>Run 916</th>
<th>Run 928</th>
<th>Run 913</th>
<th>Run 927</th>
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<tbody>
<tr>
<td>Area *1 %</td>
<td>200 *2</td>
<td>200 *2</td>
<td>50 *3</td>
<td>50 *3</td>
<td>15 *3</td>
<td>15 *3</td>
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<tr>
<td>Configuration</td>
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<td>Nozzle</td>
<td>Orifice</td>
<td>Nozzle</td>
<td>Orifice</td>
<td>Nozzle</td>
</tr>
<tr>
<td>Measured initial conditions</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Steam dome pressure MPa</td>
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<td>7.32</td>
<td>7.36</td>
<td>7.3</td>
<td>7.35</td>
</tr>
<tr>
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<td>10.5</td>
<td>11.2</td>
<td>9.4</td>
<td>11.0</td>
<td>10.1</td>
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<tr>
<td>Total jet pump discharge flow rate kg/s</td>
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<td>13.6</td>
<td>16.5</td>
<td>16.4</td>
<td>16.4</td>
<td>16.4</td>
</tr>
<tr>
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<td>20.9</td>
<td>14.2</td>
<td>17.2</td>
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<td>3962</td>
<td>3963</td>
<td>3962</td>
<td>3964</td>
<td>3961</td>
</tr>
</tbody>
</table>

*1 Break flow area of 100% corresponds to the scaled flow area of the recirculation line piping of a reference BWR.
*2 double-ended break
*3 split break
*4 Total jet pump discharge flow rate includes core inlet flow rate and bypass flow rate. Bypass flow rate is estimated to be 10% of total jet pump discharge flow rate.

### Table 12.2  Chronology of major event timings

<table>
<thead>
<tr>
<th>Event</th>
<th>Timing s</th>
<th>Run 904</th>
<th>Run 964</th>
<th>Run 916</th>
<th>Run 928</th>
<th>Run 913</th>
<th>Run 927</th>
</tr>
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<tbody>
<tr>
<td>Break</td>
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<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
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<tr>
<td>L2 * level</td>
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<td>3.0</td>
<td>3.7</td>
<td>5.6</td>
<td>8.4</td>
<td>13.4</td>
<td>17.4</td>
</tr>
<tr>
<td>L1 * level</td>
<td></td>
<td>7.1</td>
<td>8.3</td>
<td>10.3</td>
<td>14.5</td>
<td>22.6</td>
<td>32</td>
</tr>
<tr>
<td>MSIV * Closure</td>
<td></td>
<td>11.0</td>
<td>12.4</td>
<td>12.2</td>
<td>16.2</td>
<td>17.5</td>
<td>21</td>
</tr>
<tr>
<td>JPSU *</td>
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<td>9.0</td>
<td>10.7</td>
<td>13.4</td>
<td>19.6</td>
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<td>54</td>
</tr>
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<td></td>
<td>12.1</td>
<td>13.9</td>
<td>17.6</td>
<td>25.8</td>
<td>48</td>
<td>78</td>
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<td>LPF *</td>
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<td>18</td>
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<td>44.6</td>
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<td>LPCS actuation</td>
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<td>LPCI actuation</td>
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<td>82</td>
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<td>186</td>
<td>326</td>
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<tr>
<td>ADS actuation</td>
<td></td>
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<td>129</td>
<td>131</td>
<td>136</td>
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</tr>
<tr>
<td>Whole core uncover</td>
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<td>83</td>
<td>105</td>
<td>120</td>
<td>198</td>
<td>226</td>
</tr>
<tr>
<td>Whole core recovery</td>
<td></td>
<td>127</td>
<td>137</td>
<td>210</td>
<td>248</td>
<td>366</td>
<td>395</td>
</tr>
</tbody>
</table>

* note L2 level : 4.76 m from pressure vessel bottom
L1 level : 4.25 m from pressure vessel bottom
MSIV : main steam line isolation valve
JPSU : jet pump suction uncover
RLU : recirculation line uncover
LPF : lower plenum flashing
FWF : feedwater flashing
Fig. 12.1 Break configurations.

Fig. 12.2 Break flow mass fluxes.
Fig. 12.3  Break flow mass fluxes through Orifice vs. pressures.

Fig. 12.4  Break flow mass fluxes through Nozzle vs. subcooling.
**Fig. 12.5** System pressures.

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Fig. 12.7 Peak cladding surface temperatures.

Fig. 12.8 Comparison of calculated system pressures and experimental data.
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13. High Temperature ECC Injection Tests

13.1 Introduction

During BWR LOCA, ECCS injects subcooled ECC into the pressure vessel to maintain the core coolability. The LPCS and the LPCI systems take ECC from the pressure suppression pool. The ROSA-III tests have assumed ECC temperature of 313 K in accordance with the assumption made in BWR safety analyses. However, the ECC temperature may take different values because of: (1) the pool temperature variation during the normal operation of a BWR, and (2) the pool temperature rise during a LOCA due to condensation of vapor discharged from the break. Hence the effects of ECC temperature on the performance of ECCSs during a LOCA is of interest.

Two tests have been conducted in the ROSA-III facility with an extremely high (393 K) ECC temperature. The two tests, Runs 941 and 940, simulated a 200% double-ended break and a 5% small break in the recirculation pump suction line, respectively. In this chapter, the results from the two tests are compared with results from comparable tests conducted for the normal (313 K) ECC temperature, to study the ECC temperature effects on the BWR thermal-hydraulic response during LOCAs.

13.2 Test Conditions

The test conditions of the two high temperature (HT) ECC injection tests, Runs 941 and 940, and the two base case tests, Runs 926 and 922, are summarized in Table 13.1. The test initial conditions were the same for all the tests. The break size was 200% for Runs 926 and 941 and 5% for Runs 922 and 940. The test boundary conditions were the same for each set of tests until the initiation of LPCS and LPCI, since the HPCS system was assumed to fail in all the tests.

The ECC temperature was 313 K for the base case tests and 393 K for the HTECC injection tests. LPCS and LPCI were initiated when the vessel pressure decreased to their respective shutoff heads of ECCS pumps in the reference BWR (2.2 MPa for LPCS and 1.65 MPa for LPCI).

The four ROSA-III core bundles were used to simulate one peak-power bundle (bundle A) and three average-power bundles (bundles B, C and D) in the reference BWR. The ratio of power per bundle between the peak- and average-power bundles was 1.4.

13.3 Test Results

The chronology of major events during the four tests is shown in Table 13.2. The sequence of events was unaffected by the ECC temperature, whereas the timings of events were changed.

The following paragraphs discuss mainly the difference between results from the HTECC injection test and the base case test. The major events and thermal-hydraulic responses during large- and small-break LOCA tests are described in detail in Chapters 3, 5 and Appendices III and IV.
13.3.1 200% Break Tests

The vessel pressure responses during the two 200% break tests are compared in Fig. 13.1 and were essentially the same until the initiation of core reflooding, at 103 s when both LPCS and LPCI had initiated. After the initiation of core reflooding, both tests depressurized more slowly than before, however, the HTECC injection test depressurized more slowly than the base case test.

The ECCS actuation timings in the two tests were the same as shown in Table 13.2, because the two tests depressurized to the ECCS set-point pressures at the same timings; LPCS initiated to spray ECC into the upper plenum at 70 s (2.2 MPa) and LPCI initiated to inject ECC into the core bypass region at 95 s (1.65 MPa). The LPCS and LPCI injection rates, Fig. 13.2, were regulated by the vessel pressure to simulate the characteristics of BWR ECCS pump head versus flow, and were smaller in the HTECC injection test because of higher vessel pressure. However, the difference was less than 16% of the base case test injection rate.

The vessel mixture levels in the two tests, measured by conductivity probes, are compared in Fig. 13.3. A mixture level was formed below the core inlet before the core became empty of liquid, as the CCFL at the core inlet limited liquid drainage from the core into the lower plenum. This CCFL at the core inlet occurred after the initiation of LPF at 17 s. The lower plenum mixture level responded closely to the vessel pressure behavior; the two pressure holdings, following the FWF at 70 s and the initiation of core reflooding at 103 s, caused the mixture level to drop. The lower plenum mixture level reflects the lower plenum vapor generation rates, due to bulk flashing of liquid as well as boiling on the structure surfaces, and thus can be used as an indicator of the vapor flow rate at the core inlet.

The core mixture level responses in the two tests were the same throughout the core emptying period, until about 120 s when the core mixture level had recovered. The core once became completely empty of liquid. The pressure hold following the FWF, at 70 s, broke the core inlet CCFL temporarily, and thus accelerated the core emptying.

The core reflooding began at 103 s, being caused primarily by the LPCI water flowing into the core inlet region through the bypass leak holes. It is interesting to note that the LPCI water was nearly saturated (within 10 K) when it reached the core, irrespective of the ECC temperature. The LPCI water temperature was raised, in the core bypass region, by the release of stored heat from the structures and direct-contact condensation of vapor on the LPCS water. The contribution of LPCS to reflooding was small in both tests, since the drainage of LPCS water into the core was stopped by CCFL at the upper tie plate (UTP), as soon as the core reflooding increased the vapor updraft velocity at the core exit.

In the base case test, the whole core was reflooded within 25 s after the initiation of reflooding, however, the top of core was uncovered thereafter intermittently, until the lower plenum was filled with two-phase mixture. In the HTECC injection test, reflooding beyond the core mid-plane in the three average power bundles B, C and D delayed in comparison with the base case test. Such difference between the two tests became considerable after about 120 s. At the same time, the lower plenum mixture level began to rise. This was not due to flashing, but resulted from drainage of liquid from the core into the lower plenum because of CCFL breakdown at the core inlet as the pressure remained almost constant. The lower plenum liquid level rise was faster in the HTECC injection test.

The above experimental results should mean that different core-inlet CCFL situations occurred in the two tests, because of different pressure responses. This led to different parallel channel behaviors of core bundles in the two tests. The core-inlet differential pressure measurements indicated that, in the base case test, the core inlet CCFL broke down in the
peak-power bundle (bundle A) and in one of the three average-power bundles (bundle D), while other two average-power bundles were flooded by vapor upflow. In the HTECC injection tests, however, all the three average-power bundles drained, and the peak-power bundle alone was flooded. Despite such difference in the peak-power bundle inlet conditions, the reflooding behavior in the peak-power bundle was much the same in the two tests.

Figure 13.4 shows the dryout and quench front behaviors on four heater rods, chosen from the four rod bundles. The rod temperature behavior was nonuniform in each bundle (not shown) as well as among the bundles. However, the overall trends, shown in Fig. 13.4, were the same for the two tests until about 120 s. The upper and lower portions of the rods were quenched in the top-down fashion, as the pressure holding following the FWF broke the CCFL at the UTP so that the LPCS water penetrated into the core. The quenched rod surface area was larger in HTECC injection test probably because of the smaller steam updraft velocity at the UTP. After the beginning of core reflooding, the quenched rod surfaces dried out again, except at the core bottom region, as the CCFL at the UTP started again. The rod surface remained dry until the core was finally reflooded. The rod surface temperature behavior was the same in the two tests, except at the locations above the core mid-plane of the average-power bundles, where quenching of the rod surface delayed in the HTECC test because of the delayed core reflooding.

The PCTs in the two tests were nearly the same as shown in Fig. 13.5, and the PCTs were recorded at the core mid-plane in the peak-power bundle, where the reflooding and the rod quenching were not affected by the ECC temperature.

13.3.2 5% Break Tests

The vessel pressure responses in the two tests are compared in Fig. 13.6. In both tests the system depressurized, after the initiation of the ADS at 162 s, exactly in the same manner until the initiation of LPCS. Thereafter, the system depressurized more slowly in the HTECC injection test than the base case test. The slower depressurization in the HTECC test delayed the initiation of LPCI by 16 s in comparison with the base case test.

The core mixture level responses in the two tests are quite similar as shown in Fig. 13.7, except for the difference in the timing of core reflooding. The delay of reflooding in the HTECC injection test corresponded exactly to the 16 s delay of LPCI actuation, relative to the base case test.

Figure 13.8 shows the rod-surface dryout and quench behavior in the two tests. In both tests, most of the rod surfaces were quenched by LPCS in the top-down fashion, and remained wet thereafter. Several fuel rods remained dry and were not quenched until the core was reflooded by LPCI, but these rods did not contribute to the PCTs in the two tests.

The top-down quench behavior was quite similar between the two tests. This should mean that the ECC temperature did not affect the CCFL behavior at the UTP at all. The CCFL characteristics may be modified if liquid is subcooled1). However, the temperature measurements above UTP showed that the LPCS water was saturated irrespective of the ECC (LPCS) temperature, because of vapor condensation by ECC in the upper plenum. The vapor updraft velocity at the core outlet, the controlling parameter to CCFL, depends on vapor generation rates in both lower plenum and core. The lower plenum vapor generation rate depends on the vessel depressurization rate, which in turn depends on the ECC temperature as has been stated before. However, for small break tests like a 5% break test, vapor generation in the lower plenum is estimated to be much smaller than that due to quenching of core, because of the small depressurization rates. Consequently, the top-down quench process in a small break LOCA test is insensitive to the ECC temperature.
The PCTs observed in the two tests were nearly the same as shown in Fig. 13.9, since the core dryout and quenching behaviors were quite similar in the two tests as has been stated.

### 13.4 Discussions

One of the important observations in the present four tests was that ECC became quickly saturated or nearly saturated when it was injected into the upper plenum and core bypass regions, before reaching the core. Thus the subcooling of ECC had no direct impact on core cooling, although it had indirect impacts by changing the inventories and spatial distributions of mass and energy in the system. This also means that the ECC temperature effects on the vessel pressure response can be evaluated from the system mass and energy balance, assuming the thermal equilibrium between the phases.

The vessel pressure is affected by the core power, the release of wall stored heat, the ECCS injection and the break flow (including the ADS flow). Thus,

$$
\frac{dP}{dt} - \left( \frac{dP}{dt} \right)_{\text{core}} + \left( \frac{dP}{dt} \right)_{\text{wall}} + \left( \frac{dP}{dt} \right)_{\text{ECCS}} + \left( \frac{dP}{dt} \right)_{\text{break}}
$$

(13.1)

When the liquid and vapour are in thermal equilibrium, the terms in the right hand side of Eq. (13.1) are

$$
\left( \frac{dP}{dt} \right)_{\text{core}} = Q_{\text{core}} \cdot \left( \frac{\partial v}{\partial h} \right)_P / \text{DOM}
$$

(13.2)

$$
\left( \frac{dP}{dt} \right)_{\text{wall}} = Q_{\text{wall}} \cdot \left( \frac{\partial v}{\partial h} \right)_P / \text{DOM}
$$

(13.3)

$$
\left( \frac{dP}{dt} \right)_{\text{ECCS}} = \left( v_{av} + (h_{ECCS} - h_{av}) \cdot \left( \frac{\partial v}{\partial h} \right)_P \right) / \text{DOM}
$$

(13.4)

$$
\left( \frac{dP}{dt} \right)_{\text{break}} = W_{\text{break}} \cdot \left( v_{av} + (h_{break} - h_{av}) \cdot \left( \frac{\partial v}{\partial h} \right)_P \right) / \text{DOM}
$$

(13.5)

where

$$
\text{DOM} = M \cdot \left( \frac{\partial v}{\partial P} \right)_h - V \cdot \left( \frac{\partial v}{\partial h} \right)_P
$$

(13.6)

$$
\bar{v} = V / M
$$

(13.7)

$$
\bar{h} = x \cdot h_G + (1 - x) \cdot h_L
$$

(13.8)

$$
x = (\bar{v} - \bar{v}_L) / (\bar{v}_G - \bar{v}_L)
$$

(13.9)

in which \( P \) = vessel pressure, \( Q \) = heat input, \( v \) = specific volume, \( h \) = enthalpy and \( W \) = mass flow rate, \( V \), \( M \) = total volume and mass in the primary system, respectively. The suffixes \( G \) and \( L \) denote saturated vapour and liquid respectively. The wall stored heat release, \( Q_{\text{wall}} \), in Eq. (13.3) can be estimated from the following system mass and energy balance equations with Eqs. (13.7) through (13.9) using the experimental values of the residual mass and energy in the primary system at the LPCS actuation, the vessel pressure, flow rates and enthalpies of ECCS and break flow, core power and the time derivative of the vessel pressure.

$$
\frac{dM}{dt} = W_{\text{ECCS}} + W_{\text{break}}
$$

(13.10)

$$
\frac{d(Mh_k)}{dt} = W_{\text{ECCS}} h_{ECCS} + W_{\text{break}} h_{break} + Q_{\text{core}} + Q_{\text{wall}} + V_{\text{total}} \frac{dP}{dt}
$$

(13.11)

In Fig. 13.10 the calculated values of Eqs. (13.2) through (13.5) are shown for the period after the LPCS actuation. The ECC injection and break flow cause depressurization, whereas the core power and the stored heat release raise the system pressure. A higher ECC
temperature results in a smaller depressurization. However, as clearly shown in this figure, direct effect of ECC temperature on the vessel pressure is compensated by the effects of break flow and stored heat release. Namely, the stored heat release is smaller and the break flow is larger with a higher ECC temperature, and vice versa. These results are consistent with the small difference in the measured depressurization rate between the base case and HTECC injection tests, especially in the large break tests.

The wall stored heat release, $Q_{\text{wall}}$, calculated from Eqs. (13.10) and (13.11) are shown in Fig. 13.11, and are greater than two times of core power after LPCS actuation. This result is peculiar to the scaled test facility, where the stored heat effects are more significant than in a full-scale system. The depressurization during the BWR LOCA is expected faster than in the ROSA-III test because of smaller structure stored heat per unit fluid volume than in the ROSA-III.

The ECC temperature affected the core mixture level only indirectly, by affecting the pressure response as has been discussed. Higher ECC temperature led to smaller depressurization rate, and thus to less vapor generation rates in the lower plenum. The less lower plenum vapor generation rates resulted in less significant core-inlet CCFL during the core reflooding in the 200% break test, whereas it did not influence the PCTs.

Regarding the core cooling performance of ECCS, another important observation in the present tests was that ECC temperature did not affect directly the CCFL characteristics at either the core inlet or outlet, because ECC was nearly saturated when it reached to these regions. This resulted in the 5% break test's core temperature response which was essentially unaffected at all by the ECC temperature difference of 80 K.

13.5 Conclusions

The effect of the raised ECC temperature on the performance of ECCSs have been studied experimentally for both large and small break LOCA situations. The conclusions derived from this study are:

- The ECC temperature had no direct influence on the core cooling, since ECC became nearly saturated before reaching the core, because of vapor condensation on, and stored heat release to ECC. However, the ECC temperature affected indirectly the ECCS core cooling performance by changing the vessel pressure response.

- A higher ECC temperature resulted in delayed core level recovery in the upper core region, because it caused a slower depressurization, and thus resulted in smaller ECCS flow rate, delayed LPCI initiation and earlier CCFL break down at the core inlet.

- The PCT occurred at the core mid-plane where the timing of rod surface quenching was essentially unaffected by the ECC temperature. The measured peak cladding temperature was not affected by the ECC temperature for both large and small break experiments.

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Table 13.1  Summary of test conditions

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<td>ECC Temperature (K)</td>
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* % of recirculation pump suction line cross section

Table 13.2  Chronology of major events

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<td>182</td>
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b) HTECC injection test (Run 941)
a) Base case test (Run 926)

b) HTECC injection test (Run 941)

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b) HTECC injection test (Run 940)

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14. Recirculation Pump Discharge Line Break Test Series

14.1 Introduction

The BWR/6 system has two recirculation loops with one main recirculation pump and twelve jet pumps per one loop. The minimum flow areas of the jet pump drive nozzles (JPDNs) and the main recirculation pump (MRP) are smaller than the recirculation loop pipe flow area. Therefore, the break location in the main recirculation loop affects the break flow and consequently the LOCA scenario. Up to the present time, various LOCA simulation experiments have been performed, however, no integral test of the recirculation pump discharge line break LOCA has been performed except for this program.

In the ROSA-III program, three recirculation pump discharge line break experiments had been performed as one series of the break location parameter tests series. Figure 14.1 shows an isometric view of the ROSA-III facility for the recirculation pump discharge line break simulation, of which broken recirculation loop was modified from the standard recirculation pump suction break configuration. The break units are located between the main recirculation pump (MRP)-2 and the jet pump (JP)-3,4 drive lines. The break areas are chosen as 200% (double-ended break), 100% (split break) and 50% (split break) of the 1/424 scaled BWR pipe flow area.

This chapter presents the experimental results with emphasis on (1) the fundamental features of the 200% recirculation pump discharge line break experiment and the effects of break area on the transient LOCA phenomena, and (2) the comparison of the recirculation pump discharge line break and the suction line break. In addition to these, the similarity between the ROSA-III test and a postulated LOCA in a BWR/6 system is also discussed.

14.2 Test Conditions and Test Procedure

The test conditions for the three discharge line break tests are summarized in Table 14.1. The primary initial test conditions simulate the rated thermal conditions of the reference BWR except for the heat flux of fuel rods and core flow rate. As the initial heat flux of the ROSA-III tests is limited to 44% of the BWR, the initial core flow rate, including a 9.5% core bypass flow, is also lowered in order to simulate the initial enthalpy distribution across the core. Due to the limited initial core power and half length of core, the mass flux in one ROSA-III bundle is 23% of that in one nuclear bundle. The initial core power is maintained at a constant value for 9 s after the break and thereafter decreases along the power curve simulating the decay power and stored heat in the nuclear rods.

The pressure control system that keeps the pressure above 6.7 MPa is available in the three tests. The MSIV closure is simulated by closing an air valve in the main steam line (MSL) upon a trip signal of L2 with time delay of 3 s. The feedwater system, which supplies hot water (489 K) to the downcomer in the initial state, terminates after the break. As the hot water (3.5 x 10^{-2} m^3) remaining in the feedwater line is not isolated from the PV, it flows into the downcomer after the flashing initiation.

The JPDNs in the broken recirculation loop have an inner diameter of 8.4 mm and total flow area of 1.11 x 10^{-4} m^2 (A_i = 21% of the 1/424 scaled BWR recirculation pipe flow area). The inner diameter and flow area of the main recirculation pump discharge nozzle
(MRPN) are 18.7 mm and $2.75 \times 10^{-4}$ m$^2$ ($A_p = 51\%$), respectively. The break geometry is simulated by the break nozzles (Fig. 14.2) with a 26.2 mm I.D. for the 200 and 100% break tests and an 18.5 mm I.D. for the 50% break test. The 200% double-ended break (DEB) test is initiated by quickly opening the two "quick opening" valves (of break units A and B) and simultaneously closing the "quick shutoff" valve, whereas the 100 and 50% split break tests are initiated by opening only one quick opening valve (of break unit B) and keeping the quick shutoff valve opened. The recirculation pumps begin to coast down after the break.

The HPCS single failure is assumed in the three tests, therefore, the LPCs and three LPCIs are actuated by a trip signal of the L1 level with a time delay of 40 s at the designed actuation pressures (see Table 14.1).

### 14.3 Primary Thermal-Hydraulic Phenomena and Effects of Break Area in Discharge Line Break Tests

In the present section, the primary thermal-hydraulic phenomena in a 200% recirculation pump discharge line break test (Run 961) are described at first, and next, the effects of break area on the primary thermal-hydraulic phenomena are presented by comparing the 50 and 100% break tests with the 200% break test.

The lower plenum pressure, a representative of the system pressure, began to decrease after the break due to the mass discharge from the two break nozzles and showed recovery at 8.5 s after the break because of the MSIV closure, as shown in Figs. 3(a) and (b). Thereafter, the lower plenum pressure was primarily controlled by the steam balance in the system, i.e., steam generation in the core that decreased at 9 s after the break due to power decay, steam discharge through the breaks due to a water level decrease below the recirculation pump suction line (22 s), steam generation due to flashing of saturated water in the lower plenum (30 s) and the feedwater line (105 s), and steam discharge by the ADS actuation (134 s). The major events of Run 961 are shown in Table 14.2. Among them, the times of MSIV closure, recirculation line uncover (RLU), and ADS actuation were determined by the downcomer water level transient. Thus, it is shown that the water level transient in the downcomer has dominating influences on the thermal-hydraulic phenomena during a recirculation line break LOCA.

Figures 14.3(a) and (b) also show large pressure drops across the JPDNs ($A_i = 21\%$) and the MRPN ($A_p = 51\%$), indicating two-phase flow choking at the nozzles after each suction line uncover, as well as the two-phase choking at the break nozzles (26.2 mm I.D. $\times$ 2, $A = 200\%$). These flow restrictions are illustrated in Fig. 14.4. Since the flow areas at the JPDNs and MRPN determined the discharge mass flow from the pressure vessel instead of the break flow areas, $A_i + A_p (= 72\%)$ is named here as the effective choking flow area in the 200% discharge line break test.

The break flow rates were obtained from the average fluid density measured by a two-beam gamma densitometer and the momentum flux measured by a drag disk. The break flow rates at both break units showed that the break flow at the jet pump side break unit began to decrease abruptly at 17 s and that at the recirculation pump side break unit began to decrease similarly at 22 s due to initiation of the steam discharge through each break unit. The cumulated water mass which discharged through the two break units before the initiation of the jet pump suction line uncover (JPSU, 17 s) was 182 kg, which agreed well with the downcomer water mass between the initial water level (DL 5.0 m, from PV bottom) and the jet pumps suction line (DL 2.8 m).

Figure 14.5 shows the mixture levels measured by the conduction probes and the
collapsed water levels estimated from differential pressures in the pressure vessel. Existence of the mixture level in the lower plenum after the lower plenum flashing (LPF), indicated the countercurrent flow limiting at the core inlet. This flow limiting contributed to lower the mass depletion from the core and to keep the collapsed water level inside the core shroud higher than that in the downcomer. The mixture level in the core was kept slightly higher than the collapsed water level in the blowdown phase, and the difference between the mixture level and the collapsed level in the core became large after the LPCS actuation.

**Figure 14.6** shows a representative transient of the fuel rod surface temperatures at the seven elevations of the corner peak power rod in high power bundle A. A dryout caused temperature excursion, which occurred after the break at the upper part of the fuel rod and was terminated by the LPF initiated at 30 s after the break. The second dryout occurred from the top to the bottom of the core, showing strong correlation with the mixture level transient in the core. The dryouts in the average power bundles occurred slightly earlier than those in the high power bundle. The reflooding began after LPCI actuation and proceeded with the collapsed water level rise. In the reflooding phase, two types of quench were observed, namely, the top-down quench by the LPCS water and the bottom-up quench by LPCI water. The quench times of fuel rods were widely distributed between the times of LPCS actuation and arrival of bottom-up quench front in both high and average power bundles. Therefore, the differences of dryout and quench phenomena among the four fuel bundles were not so significant as the distribution in each fuel bundle. The PCT of 894 K (621 °C) was observed at the highest power location in the high-power bundle (position 4 of rod A-82) at 155 s after break (11 s after LPCI actuation).

The results of 100% (Run 963) and 50% (Run 962) breaks are compared with those of 200% break (Run 961) and the effects of break area on the pressure response, fuel rod temperatures, and mass inventory inside core shroud are shown below.

**Figure 14.7** shows the lower plenum pressures and highest rod surface temperatures in the three tests. It is found from this figure that the pressure response of the 100% break test agreed well with that of a 200% break. This agreement was ascribable to the same effective choking flow area (\(A_{j} + A_{p} = 72\%\)) at the JPDNs and the MRPN between the two tests. A slight pressure difference occurred between the JPSU time (17 s) of the 200% break test and the RLU time (25 s) of the 100% break test due to the different downcomer water level falling speeds between the two tests. The large difference of volumetric flow rates between the two-phase flow after the JPSU in the 200% break test and the single-phase flow before the RLU in the 100% break test is the main reason for the pressure difference between them.

On the other hand, two-phase flow choking in the 50% break test occurred only at the break nozzle (18.5 mm l.d.), which was confirmed from the pressure distribution along the break flow path. The maximum pressure difference across the JPDNs and MRPN was below 0.6 MPa even after the recirculation line recovery. In the 50% break test, the flow resistance at these nozzles contributed little to the break flow compared with the flow choking at the break nozzle, and the effective choking flow area was not the JPDN and the MRPN areas, but the break nozzle flow area of 50%. The difference of the effective choking flow areas between the 200 and 50% break tests caused the difference of depressurization rates after the RLU.

The fuel rod surface temperatures in the three tests showed similar trends, however, the PCT became higher for the smaller break. The reason for the highest PCT in the 50% break test is as follows. The differences in the dryout initiation times of the fuel rods were small (less than 16 s) between the 50 and 200% break tests, whereas the actuation time of the LPCI in the 50% break test differed by 56 s from that of the 200% break test. The delayed
14.4 Comparison between Discharge and Suction Line Breaks

The suction line break tests were performed previously with test conditions similar to those of the discharge line break tests, except for the break area and break location. Representative test results, such as pressure response and fuel rod surface temperature in the 200% discharge line break test, are compared with those in the 75% suction line break test\(^9\) (Run 929) in Fig. 14.8, and major test events of these two tests are compared in Table 14.2. The effective choking flow area in the 75% suction line break test was the break flow area, which was close to the effective choking flow area of 72% in the 200% discharge line break test. Agreement between the two tests was the results of similar effective choking flow areas between them. Similarly, the test results of the 50% discharge and suction line break\(^9\) (Run 928) tests shown in Fig. 14.9 also indicated the analogy of major events between these two tests. Thus, an analogy between the recirculation pump discharge line breaks and suction line breaks has been confirmed as shown in Table 14.3.

From the above discussion, it is expected that a discharge line break with any break area can be related to a suction line break. Namely, test results of a discharge line break with a break area larger than 72% (= \(A_d + A_r\)) of the scaled recirculation line flow area agree well with those of a 72% suction line break and discharge line break with a break area less than 72% agree well with those of a suction line break with the same break area.

Note that the maximum effective choking flow area for the BWR discharge line breaks is different from the 72% of the ROSA-III test because the least flow area at the MRPN is different from plant to plant (from 30 to 100% of the scaled recirculation line flow area). The conclusions obtained above, however, can be applied to the BWR system if the 72% of the ROSA-III effective choking flow area is replaced by the generalized form of effective choking flow area or \((A_d + A_r)\) for the BWR/6 system.

14.5 Similarity Analysis of ROSA-III Test and BWR/LOCA

The ROSA-III 200% discharge line break test and the same LOCA in the BWR/6 were analyzed using the same analytical models to clarify their similarity. Analyses were performed using the RELAP5/MOD1\(^{10}\) (cycle 18) code with an improved jet pump model\(^{11}\) developed at JAERI.

The ROSA-III system was nodalized by 64 volumes, 72 junctions, and 26 heat structures for the case R calculation. The three average power bundles were combined into a pipe component (C40), and the high power bundle was represented by a pipe component (C45). The heat structures were considered for the heater rods and the pressure vessel wall. Two types of heater rods were calculated in the high power bundle: heat structures with radial \times local peaking factor of 1.4 \times 1.13 for peak power rods and 1.4 \times 0.938 for the other rods. The heater rods in the average power bundle had radial \times local peaking factors of 1.0 \times 1.0. Momentum mixing in the jet pump was calculated for a single-phase flow, and the jet pump model was replaced by a branch model included in the code for a two-phase flow. A separator model was used for the steam separator (C61). A nonequilibrium calculation model was used in each component. The recirculation pump performance in a single-phase water flow
was simulated in the calculation and that in a two-phase flow was simulated by the semiscale pump characteristics included in the code. Such geometric data as volume, length, area, form loss, hydraulic equivalent diameter, and elevation change were modeled accurately for the ROSA-III facility. The initial and transient test conditions were also well simulated in the analysis.

Analysis was also performed for the 200% discharge line break LOCA in the BWR/6 system with a single failure of HPCS (case B). Analytical conditions of case B are the same as those of case R except for the differences in the volumetric scale, height and configuration of components, initial core flow, and initial rod power. The Bingham pump model was applied to the MRP in case B.

Calculated results for ROSA-III test Run 961 (case R) are compared at first with the experimental data (ROSA-III EXP) for the evaluation of the analysis models, and we conclude that the analysis models and conditions for the ROSA-III discharge line break test are sufficiently correct for the prediction of major events.

Figures 14.10(a) and (b) show the pressure responses and fuel rod surface temperatures at the center of average power rods (peaking factors of $1.0 \times 1.0$) for the case R, case B and ROSA-III EXP. Figure 14.11 show the fuel rod surface temperatures of maximum power (peaking factors of $1.4\times1.13$) rod for the three cases. Figure 14.10(a) also shows a good coincidence of system pressure responses between the two analyses, case R and case B. This coincidence means the same time scale for the key events, such as initiation of LPF and ECCS actuation between the ROSA-III and BWR systems. The slight difference in pressure response in the early blowdown phase results from different MSIV trip logics. The MSIV in case R was tripped at 8.5 s after the break, simulating the experimental result with the trip level of L2 + 3 s, while the MSIV in case B was tripped 13 s after the initiation of the pressure control system and had a lesser effect on the system pressure response. The total discharged mass from the breaks of case R, however, agreed well with the 1/424 scaled mass of case B during the 180 s after the break. The reason of similar responses in pressure and mass inventory between the ROSA-III and the reference BWR is mainly attributed to the volumetric scaling concept on the fluid volume in each component; the choking flow areas, such as the jet pump flow area, MRPN flow area, and break flow area; and the overall power generation rate.

The initiation of a temperature rise due to dryout of fuel rods in case R and case B agreed well. The agreement of dryout initiation times between case R and case B is important for similarity evaluation. The surface temperature responses of fuel rods of case R, however, did not show a higher temperature rise in the early blowdown phase. On the contrary, they showed slightly higher temperatures in the later blowdown phase compared with those of case B, as shown in Figs. 14.10(b) and 14.11.

The early dryout of the maximum power rods in the central BWR core was calculated at 1.2 s after break due to rapidly decreased critical heat flux (CHF), which was caused by relatively faster core flow cooldown and void increase, compared with a slower decrease of fuel rod surface heat flux (Fig. 14.12). Also, surface heat flux of the BWR rods was higher than that of the ROSA-III rods in the early blowdown phase due to the limited power generation rate for 9 s after break in the ROSA-III core. Therefore, the maximum power rod of ROSA-III did not show a temperature rise due to lower surface heat flux compared with that of the BWR rod. On the other hand, the average power rods of both ROSA-III and the BWR showed similar temperature responses in the early blowdown phase because their CHFs were much higher than their surface heat flux.

Through these comparisons we conclude that system pressure, break flow rates, mass
inventory and key events in the discharge line 200% break show a similarity between ROSA-III and the reference BWR. The rod surface temperature of the ROSA-III test shows conservatism in the later blowdown phase due to a slightly higher heat flux. However, the surface temperature of the ROSA-III maximum power rods shows a lower value in the early blowdown phase due to limited power compared with those of a BWR.

14.6 Conclusions

The following conclusions were obtained from the experimental results on recirculation pump discharge line break tests at ROSA-III with break areas between 50 and 200% and from the similarity analysis between the ROSA-III test and a postulated LOCA in a BWR/6 using the RELAP5/Mod1 code with the JAERI jet pump model.

(1) The transient thermal-hydraulic behaviors of the discharge line break are experimentally confirmed to be similar to the suction line break by using the effective choking flow area that dominates the transient LOCA phenomena. Namely, a discharge line break with a break area larger than \( (A_4 + A_p) \) is similar to a suction line break with a break area of \( (A_2 - A_r) \), where \( A_4 \) and \( A_p \) are the choking flow areas at the JPDNs and MRPN, respectively. Discharge and suction line breaks with the same break areas smaller than \( (A_4 + A_p) \) are similar. In the ROSA-III facility \( A_4 \) and \( A_p \) are 21 and 51% of the 1/424 scaled BWR recirculation line flow area \( (A_s) \), respectively.

(2) The maximum effective choking flow area for the recirculation discharge line breaks is \( A_4 + A_p \) and that for the suction line breaks is \( (A_2 + A_r) \).

(3) Good similarity is confirmed through the analyses between the ROSA-III test and a BWR discharge line 200% break LOCA. The conclusions mentioned above can be applied to the BWR LOCA phenomena by using the generalized form of the effective choking flow area.

(4) The analysis showed that the ROSA-III heater rod surface temperature in the early phase of blowdown was atypical due to the limitation of the ROSA-III electric power supply system for the core heater rods. The maximum power rod of a BWR showed a larger temperature rise than that in the ROSA-III test due to the DNB caused by a higher heat flux and faster flow coastdown appearing shortly after the break in the BWR core.

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### Table 14.1  Test conditions of discharge line break tests

<table>
<thead>
<tr>
<th>Break Condition</th>
<th>Run 961</th>
<th>Run 963</th>
<th>Run 962</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break Area (%)</td>
<td>200</td>
<td>100</td>
<td>50</td>
</tr>
<tr>
<td>Mode</td>
<td>DEB</td>
<td>Split</td>
<td>Split</td>
</tr>
<tr>
<td>Initial Condition</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure (MPa)</td>
<td>7.35</td>
<td>7.35</td>
<td>7.35</td>
</tr>
<tr>
<td>Core Inlet Subcooling (K)</td>
<td>11.0</td>
<td>10.4</td>
<td>10.9</td>
</tr>
<tr>
<td>Upper Plenum Quality (%)</td>
<td>13.0</td>
<td>13.1</td>
<td>13.2</td>
</tr>
<tr>
<td>Core Power (MW)</td>
<td>3.98</td>
<td>3.97</td>
<td>3.97</td>
</tr>
<tr>
<td>Water Level (m)</td>
<td>5.0</td>
<td>5.0</td>
<td>5.0</td>
</tr>
</tbody>
</table>

**ECCS Injection Logics**

- **LPCS**: 11 + 40 s and \( P \leq 2.2 \) MPa
- **LPCI**: 11 + 40 s and \( P \leq 1.6 \) MPa

### Table 14.2  Comparison of major events between discharge and suction breaks

<table>
<thead>
<tr>
<th>Events</th>
<th>DISB Run 961</th>
<th>SUCB Run 929</th>
<th>DISB Run 962</th>
<th>SUCB Run 928</th>
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</thead>
<tbody>
<tr>
<td>Break Initiation</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>Feedwater Line Closure</td>
<td>1.7</td>
<td>2.0</td>
<td>1.7</td>
<td>1.9</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>8.5</td>
<td>6.6</td>
<td>12.0</td>
<td>14.0</td>
</tr>
<tr>
<td>JP Suction Uncovering</td>
<td>17.0</td>
<td>12.0</td>
<td>24.0</td>
<td>20.0</td>
</tr>
<tr>
<td>Rec. Line Uncovering</td>
<td>22.0</td>
<td>14.0</td>
<td>32.0</td>
<td>24.0</td>
</tr>
<tr>
<td>Lower Plenum Flashing</td>
<td>30.0</td>
<td>27.0</td>
<td>48.0</td>
<td>45.0</td>
</tr>
<tr>
<td>Dryout at Top of Core</td>
<td>46.0</td>
<td>42.0</td>
<td>62.0</td>
<td>68.0</td>
</tr>
<tr>
<td>Dryout at Bottom of Core</td>
<td>81.0</td>
<td>89.0</td>
<td>116.0</td>
<td>120.0</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>105.0</td>
<td>108.0</td>
<td>167.0</td>
<td>148.0</td>
</tr>
<tr>
<td>ADS Actuation</td>
<td>134.0</td>
<td>131.0</td>
<td>139.0</td>
<td>137.0</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>144.0</td>
<td>145.0</td>
<td>200.0</td>
<td>188.0</td>
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<tr>
<td>Completion of Quench</td>
<td>215.0</td>
<td>212.0</td>
<td>272.0</td>
<td>245.0</td>
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<tr>
<td>Break Area (%)</td>
<td>200</td>
<td>75</td>
<td>50</td>
<td>50</td>
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</table>
Table 14.3 Identification of ROSA-III recirculation line breaks on a viewpoint of effective choking flow area

<table>
<thead>
<tr>
<th></th>
<th>Effective Choking Flow Area (%)</th>
</tr>
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<tbody>
<tr>
<td></td>
<td>121</td>
</tr>
<tr>
<td>Disch. L. Breaks</td>
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<tr>
<td>Run No.</td>
<td></td>
</tr>
<tr>
<td>Break Area (%)</td>
<td></td>
</tr>
<tr>
<td>PCT (K)</td>
<td></td>
</tr>
<tr>
<td>E.C. Location *</td>
<td></td>
</tr>
<tr>
<td></td>
<td>961</td>
</tr>
<tr>
<td>Corresponding</td>
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</tr>
<tr>
<td>Suction L. Breaks</td>
<td></td>
</tr>
<tr>
<td>Run No.</td>
<td></td>
</tr>
<tr>
<td>Break Area (%)</td>
<td></td>
</tr>
<tr>
<td>PCT (K)</td>
<td></td>
</tr>
<tr>
<td>E.C. Location **</td>
<td></td>
</tr>
<tr>
<td></td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>894</td>
</tr>
<tr>
<td></td>
<td>JPDN**</td>
</tr>
<tr>
<td></td>
<td>MRPN***</td>
</tr>
<tr>
<td></td>
<td>926</td>
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<tr>
<td></td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>785</td>
</tr>
<tr>
<td></td>
<td>JPDN</td>
</tr>
</tbody>
</table>

* Effective Choking Location
** Jet Pump Drive Nozzles
*** MRP Discharge Nozzle
Fig. 14.1  Overview of ROSA-III system.

Fig. 14.2  Details of break nozzles.

<table>
<thead>
<tr>
<th>Break Area Ratio (%)</th>
<th>Break Diameter (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>200, 100</td>
<td>26.2</td>
</tr>
<tr>
<td>50</td>
<td>18.5</td>
</tr>
</tbody>
</table>

Material SUS304
Dimension in mm

Octagonal Ring R35

FLOW DIRECTION

Steam line
Pressure vessel
Broken loop
Intact loop
Feedwater line
Jet pumps
Quick Shut Valve
Recirculation Pump
Recirculation Pump
Fig. 14.3(a) Pressure distribution along jet pump side break flow path.

Fig. 14.3(b) Pressure distribution along MRP side break path.

Fig. 14.4 Flow restriction in broken recirculation loop (Run 961).
Fig. 14.5  Water levels in pressure vessel (Run 961).

Fig. 14.6  Dryout and quench of fuel rod. All related with mixture level in core (Run 961).
Fig. 14.7 Effect of break area on pressure and PCT in discharge line break tests (Runs 961, 963 and 962).

Fig. 14.8 Analogy of pressure and rod temperature responses between the 200% discharge line break (DISB) and the 75% suction line break (SUCB).

Fig. 14.9 Analogy of pressure and rod temperature responses between the DISB and SUCB with break areas of 50%.
**Fig. 10.10(a)** Comparison of pressure transients among the ROSA-III experiments, analysis (case R) and BWR analysis (case B).

**Fig. 14.10(b)** Comparison of average-power rod surface temperatures among the ROSA-III experiment, analysis (case R) and BWR analysis (case B).
Fig. 14.11  Comparison of maximum-power rod surface temperature among the ROSA-III experiment, analysis (case R) and BWR analysis (case B).

Fig. 14.12  Transient thermal-hydraulic condition in early blowdown phase of case R and case B: (a) surface heat flux, (b) average void fraction and (c) core inlet flow rate.
17. Core Heat Transfer Analysis

17.1 Introduction

The temperature behavior of a fuel rod cladding is one of the most important parameters in the LOCA analysis because it becomes difficult to maintain the integrity of fuel cladding at the high cladding temperature. Heat transfer between the cladding surface and coolant directly affects the temperature behavior of fuel rods. Therefore, the heat transfer behavior in the ROSA-III core should be carefully investigated. The ROSA-III data are considered to be useful both to assess a LOCA analysis code and to understand the mechanisms of core heat transfer during a LOCA transient of a BWR because previous works have shown that the similarity of basic thermal hydraulic phenomena during a LOCA is good between the ROSA-III and the BWR\(^{3,2,3,4}\). The ROSA-III data are considered to be important also as heat transfer data because there are few data on the heat transfer coefficients in the core obtained from large-scale experiments simulating LOCAs in the BWR.

Detailed core heat transfer analysis was conducted to understand the heat transfer behavior in the ROSA-III core during a LOCA test\(^{16}\). A finite difference method was used to obtain the heat transfer coefficient and temperature profile in the rod in the radial direction. Then the heat transfer model in the RELAP4/MOD6/U4/J3 code\(^5\) was revised using the analysis results. In this chapter the core heat transfer characteristics, post-CHF heat transfer coefficients, quench temperature and the calculation results by the original and the revised RELAP4/MOD6/U4/J3 codes are presented.

17.2 Test Conditions

Four loss-of-coolant experiments in the break area parameter test series\(^6\) are analyzed in this chapter. In this test series, the break location was fixed at the main recirculation pump suction line, and the high pressure core spray (HPCS) failure was assumed. The four tests are Run 926 with a 200% break area, Run 916 with 50%, Run 913 with 15% and Run 912 with 5%\(^7,8,9\). Major experimental conditions of each test are listed in Table 17.1.

17.3 Test Results

17.3.1 Pressure and Core Mixture Level Transient

Figure 17.1 shows the system pressure transients in the 5, 15, 20 and 200% break tests. Pressures started to decrease rapidly immediately after the break due to fluid discharge through the main steam line (MSL) and the break, and then increased temporarily after the main steam line isolation valve (MSIV) closure. The pressure in the 5% break test was kept at 8.4 MPa by the safety relief valve actuation. After the downcomer water level decreased to the level of outlet nozzle to recirculation line (recirculation line uncover; RLU), steam began to discharge from the system through the recirculation pump suction line. Therefore, the pressures began to decrease rapidly after RLU, in the 50 and 200% break tests. In the 5 and 15% break tests, a rapid decrease in the pressure occurred after ADS actuation. The ADS flow area corresponds to a 34% break area. When the pressure decreased to the saturation pressure of the fluid in the lower plenum, the fluid started flashing (LPF), which improved
core cooling and decreased the depressurization rate. The LPCS was actuated at the pressure of 2.2 MPa and LPC1 at 1.6 MPa. Chronologies of major events of the four experiments are compared in Table 17.2.

Figure 17.2 shows the steam-water mixture level in the high power bundle. Whole core was exposed to steam after LPF and the core was covered again by the steam-water mixture after ECCS actuation in every test. The mixture level behavior was basically one-dimensional among the four bundles, which results in almost the same initiation timings of core dryout and reflooding among the bundles. Similar behavior was also observed in the ESTA-II facility which is a refill-reflood simulator with the same core height and radius as the BWR to investigate multichannel effect in the BWR. In the 50 and 200% break tests, the core mixture level was recovered to the top before complete recovery of mixture level in the lower plenum.

17.3.2 Cladding Surface Temperature

Figure 17.3 shows the cladding surface temperatures of peak power rods at the midplane of core. The temperatures began to rise rapidly when the rod surface was uncovered above the mixture level. Dryout and bottom-up quenching behavior was basically one-dimensional in the core due to the one-dimensional mixture level behavior with the exception of region wetted by the top-down quench, which is occurred uniformly from top of the core. The figure shows that the cladding surface temperature behavior in the dryout period can be divided into three periods corresponding to different temperature change rates. The first period is from dryout to LPCS actuation. The second period is from LPCS actuation to reflooding initiation. The third period is from reflooding initiation to quench. The core is estimated to be cooled mainly by steam flow in the first period, by mist flow in the second period and by two-phase flow in the third period. The heat transfer behavior in each period is analyzed and characterized hereafter.

17.4 Heat Transfer Analysis Method

The ROSA-II heater rod consists of four material regions: sintered boron nitride (BN), Nichrome, packed BN and Inconel 600 as shown in Fig. 17.4. BN used for electrical insulation material has almost the same volumetric specific heat as UO₂. Nichrome is used for heater element material. Inconel 600 is a cladding material.

The temperature profile in the radial direction in the heater rod is calculated to obtain the heat flux at cladding surface and the heat transfer coefficient. When the heat flux in the axial direction can be neglected comparing with that in the radial direction, the basic equation to obtain the temperature profile in the radial direction is as follows:

\[ \rho C_p \frac{dT}{dt} - \frac{1}{r} \frac{d}{dr} \left( r h \frac{dT}{dr} \right) = Q \]

where \( C_p \) is specific heat, \( \rho \) is density, \( T \) is temperature, \( h \) is thermal conductivity, and \( Q \) is heat generation rate per unit volume. A finite difference method is used to calculate the temperature profile in the radial direction of the heater rod. The analysis method is almost identical to that stated in Ref. 11. Experimental data of the electrical power input and the cladding surface temperature were used for the calculation. The dependence of thermal conductivity and specific heat on temperature is considered in the calculation. In the finite difference calculation, the heater rod is divided into twenty meshes in the radial direction. The surface heat flux is obtained from the temperature gradient at the surface, which is a sum of convective and radiative heat flux. Heat transfer coefficients are defined by the
following equations:

\[ h_t = \frac{q_t}{T_w - T_t}, \]

\[ h_r = \frac{q_r}{T_w - T_t}, \]

\[ h_c = h_t - h_r. \]

where \( q_t \) is total heat flux calculated by temperature gradient at cladding surface, \( q_r \) is radiative heat flux, \( h_t \) is total heat transfer coefficient, \( h_r \) is radiative heat transfer coefficient, \( h_c \) is convective heat transfer coefficient, \( T_w \) is surface temperature, and \( T_r \) is saturation temperature.

The radiative heat flux is calculated by the following equation\(^1\):\(^2\)

\[ q_r = \frac{1}{\varepsilon_w + \frac{1}{\alpha_x} - 1} (T_w^4 - T_r^4) \]

where \( \alpha \) is Stefan-Boltzmann constant, \( \varepsilon_w \) is emissivity of cladding surface, and \( \alpha_x \) is absorption coefficient of steam.

An emissivity of 0.7 is used for cladding surface in the analysis, which is the value for Inconel 600 without an oxide layer on the surface. The data in Ref. 13 are used for the absorption coefficient of steam.

### 17.5 Results and Discussion of Heat Transfer Analysis

Our primary concern for the heat transfer analysis is a peak cladding temperature which was observed at the midplane in the high power bundle during the 5, 15, 50 and 200% break experiments. Therefore the results at the midplane in the high power bundle are presented hereafter.

#### 17.5.1 Steam Cooling Period (Dryout to LPCS Actuation)

Figure 17.5 shows the convective heat transfer coefficients vs. time from dryout to LPCS actuation in each test. They are the averages of heat transfer coefficients at the midplane of thirty heater rods in the high power bundle. The standard deviations of the heat transfer coefficients are also shown in the figure. The figure shows relatively high values until 10 s after dryout possibly due to enhancement of heat transfer by liquid droplets entrained in the steam flow near the mixture level. Heat transfer enhancement near the mixture level is also observed in Ref. 14. Heat transfer after 10 s is considered to be due to steam cooling. The core is cooled by steam flow generated mainly in the lower plenum because most of coolant in the system is in the lower plenum during this period. The duration of the steam cooling period is very short in the 200% test as shown in the figure. The convective heat transfer coefficients in the 5, 15 and 50% break tests are almost the same and decreased with time from 120 to 20 W/m\(^2\)K. Radiative heat transfer coefficients increased with increase in surface temperature and reached a value of 15 W/m\(^2\)K at LPCS actuation time in the 5% break test.

Figure 17.6 shows dependence of the convective heat transfer coefficients on the system pressure between dryout and LPCS actuation in 5, 15 and 50% break tests. The convective heat transfer coefficients in the figure are almost the same among the three tests. This indicates that steam flow rates in the core in this period are almost the same in the three tests. The core steam flow rate is directly affected by the discharge flow rate from the system. The
steam is discharged from the system through the break and the ADS line area. The steam discharge flow rate is dependent on the system pressure and the total discharge area which is a sum of break area and the ADS line area. The ADS line orifice area corresponds to 34% of scaled (1/424) cross-sectional area of recirculation pump suction line piping of the BWR. Dryout occurred after ADS actuation in the 5 and 15% break tests. The total discharge areas in this period are, therefore, 39 and 49% in the 5 and 15% break tests, respectively. In the 50% break test, the ADS actuation occurred at a system pressure of 2.7 MPa. Thus total discharge areas of the 5, 15 and 50% break tests are almost the same between dryout and ADS actuation in the 50% test, which result in almost the same discharge flow rate from the system when the pressure is same. The core steam flow rate is also affected by the flow path of coolant from the lower plenum to atmosphere. Two flow paths are considered, through the core and through the jet pump. The latter flow path has a smaller flow area and larger flow resistance than those of the former flow path. Thus most of the steam generated in the lower plenum discharges through the core to atmosphere. The convective heat transfer coefficients were very similar in the steam cooling period in the 5, 15 and 50% break tests because nearly the same amount of steam flowed upward through the core.

17.5.2 Spray Cooling Period (LPCS Actuation to Reflooding Initiation)

Figure 17.1 shows the convective heat transfer coefficients in the spray cooling period from LPCS actuation to reflooding initiation. The heat transfer coefficients are the averages of values at dried heater rod surfaces at the midplane in the high power bundle. Standard deviations are also shown in Fig. 17.7. The standard deviations in this period are relatively large comparing to those in the steam cooling and reflooding periods, which indicates the two-dimensional nature of the mist cooling. The PCT was observed at the heater rod which was not cooled enough by the mist flow for each experiment. The heat transfer coefficient increased from 20 to 70 W/m²K with time increase and became almost the same value in the 5, 15 and 50% break tests. It should be noted that the heat transfer coefficients in the 200% tests are approximately 70 W/m²K and change little with time as in the other tests though the steam flow rate from the lower plenum is considered higher than in the other tests due to larger break area. The value of 70 W/m²K is almost three times higher than the value measured by Naitoh at lower pressure as shown in the figure.

Figure 17.8 shows fractions of dried heater rods among thirty instrumented rods vs. dimensionless time at the midplane in the high power bundle in each test. All the heater rods at the core midplane were dried out during this period in the 50 and 200% break tests. Almost half of heater rods were cooled down by top-down quench before reflooding initiation in the 5 and 15% break tests.

Figure 17.9 shows the time variation of the ratio of heat transfer rate to coolant to power input in the 5, 15, 50, and 200% break tests. The ratio of 1.0 means that total input power to the heater rod is transferred to coolant, therefore, there is no increase in heater surface temperature. The figure shows the effectiveness of LPCS for core cooling in the test with break area of less than 50%. But the heater rods are not cooled by LPCS enough to halt the temperature increase before the reflooding initiation for the 200% break test.

17.5.3 Reflooding Period (Reflooding Initiation to Rod Quench)

Figure 17.10 shows the total heat transfer coefficients in the reflooding period. They are the average values of dried out heater rods at the midplane in the high power bundle. The figure shows different behavior between small break tests of 5 and 15%, and large break tests of 50 and 200%.
In the 5 and 15% break tests, the heat transfer coefficients increased rapidly from 80 to 450 W/m²K as shown in Fig. 17.10. These increase timings corresponded to recovery timings of the steam-water mixture level, measured by conduction probes, to the midplane of core. This indicates that the heat transfer above the mixture level by steam or dispersed flow cooling is relatively small in this period for the small break tests.

In the 50 and 200% break tests the heat transfer coefficients increased continuously with time. The lower plenum was not completely filled with water at the beginning of reflood, because the fall of water from the core inlet region to the lower plenum was limited by counter current flow limiting (CCFL) at the core inlet side entry orifice by the upward steam flow from the lower plenum. Conduction probe signals showed liquid was entrained in the steam flow above the core mixture level in a large break test. This liquid entrainment enhanced the heat transfer above the mixture level. This entrainment is considered as a characteristic phenomenon in the large break test not observed in the small break test.

It is not clearly understood about the applicability of the heat transfer coefficient data in the mist cooling and reflooding periods due to lack of local measurements in the facility and available study on this problem. However these data are considered to be very important for the code assessment or improvement because they are obtained under nearly the same condition of pressure, ECCS injection flow rate and geometry as those of the BWR.

17.5.4 Quench Behavior

Two kinds of quench were observed in the ROSA-III experiment, top-down quench and bottom-up quench. As mentioned in the previous section, top-down quench becomes prominent with decrease in break area. However top-down quench behavior showed a large radial distribution at the midplane of the core. All the rods were not quenched from the core top. The PCT was observed at the rod which was quenched from the bottom.

Figure 17.11 shows the dependence of bottom-up quench superheat on the system pressure at the midplane of core in the high power bundle for the 50 and 200% break tests. The quench superheat shows little dependence on the system pressures. The following empirical correlation was obtained from this figure.

\[ T_q = T_{sat} + 262 \]

where \( T_q \) is quench temperature in \( K \), and \( T_{sat} \) is saturation temperature in \( K \).

The data is correlated by this equation with the standard deviation of 30 K. This value appears to be higher than the existing data, which is probably due to the oxidation at the cladding surface. The bottom-up quench temperature may be lower under the BWR LOCA condition. But this uncertainty about the quench temperature prediction has no effect on prediction of the PCT.


The RELAP4/MOD6/U4/J3 code is a revised version of the RELAP4/MOD6 by Japan Atomic Energy Research Institute (JAERI)\(^5\). The major improvements and modifications included in this version were made on core heat transfer calculation after core spray initiation during a BWR LOCA. The improvements made were ; (1) to add a model for top-down quenching caused by liquid film falling-back from the upper plenum to the core, and (2) simplify the heat transfer calculation after core spray actuation to shorten the computing time. In this simplified heat transfer model, the core is vertically divided into three regions after HPCS or LPCS actuation, which are (i) a film boiling region, (ii) a region cooled by
mist flow, and (iii) a region wetted by failing liquid. The heat transfer coefficient in each region are recommended in the code manual\(^5\), and shown in Table 17.3. The film boiling region is determined by the local quality. When the local quality is less than the critical value shown in Table 17.3, the heat transfer mode is regarded as the film boiling. Only one kind of a heat transfer coefficient is given for the film boiling region in the original model.

By using the analytical results of heat transfer described in Section 17.5, the simplified heat transfer model after spray actuation has been improved with the recommended heat transfer data given in Table 17.3. The improvement is made by adding a model for bottom-up quench. In the new model, when the surface temperature of the rod under the film boiling condition becomes less than the quench temperature, the heat transfer coefficient is changed to the value for the wetted rod. Bottom-up quench behavior can be better calculated by using this new model than the original one. The recommended heat transfer coefficients and the quench temperature are listed in Table 17.3. The recommended values for the mist cooling and film boiling heat transfer coefficients are obtained by averaging the heat transfer data at the midplane in the high power bundle shown in Figs. 17.7 and 17.10, respectively.

Effect of steam superheat in the core on the cladding surface temperature behavior is not considered in the calculation model. The heat transfer coefficient is defined by using the saturation temperature. When the constant heat transfer coefficient obtained at the midplane is used throughout the core in the simplified heat transfer model, the temperature difference between the cladding surface and steam, and hence heat transfer rate become overpredicted in the upper part of the core, and those become underpredicted in the lower part. This effect will be checked by the comparison of calculated and experimental results hereafter.

The 200% break tests, Run 926 was analyzed using the original and the new models. Noding of the system and model options are almost the same as described in Ref. 2. The simplified heat transfer model is used after LPCS actuation. The heat transfer data used in the two analyses are listed in Table 17.3. The calculated results showed that the thermal-hydraulic behavior is similar for both of the two analyzed cases except for the rod surface temperature transients. Figure 17.12 shows the comparison of the rod surface temperatures in the 200% break test and the calculated results using the original and the new models. Dryout timings after lower plenum flashing are earlier in the two calculations than test data because the critical heat flux conditions are not calculated well. Top-down quenching is calculated at an elevation of 1.9 m in Fig. 17.12 using the top-down quenching model. Experimental data at this elevation shown in the figure was not quenched from the core top, but some heater rods were locally wetted at this elevation from the core top after LPCS actuation. This two-dimensional behavior of top-down quenching can not be calculated by this one-dimensional top-down quenching model. Heater rods are not quenched at a level lower than 1.4 m in the original case because of absence of a bottom-up quench model. The calculated temperature increasing rates after LPCS actuation in the original case are larger than the test data. This indicates the underestimation of the heat transfer coefficients for the spray cooling and the film boiling in the original case. By using the present model and the data, it is shown at the lower three elevations in the figure that the surface temperature behavior is predicted well after LPCS actuation until the bottom-up quench, although the heat transfer model in the code is very simple. This also indicates the effect of steam superheat in the core is not significant. Figure 17.12 demonstrates the usefulness of the simplified heat transfer model when the heat transfer data is properly chosen.
17.7 Conclusions

Following conclusions were obtained through a detailed experimental analysis of core heat transfer during the four ROSA-III LOCA experiments with different break areas and the analysis of data by using the revised RELAP4/MOD6/U4/J3 code.

1) The core dryout period can be divided into three: steam cooling, spray cooling and reflooding periods. Core heat transfer behavior was basically one-dimensional during the steam cooling and reflooding periods. This is because of one-dimensional two-phase mixture level behavior among the bundles.

2) The convective heat transfer coefficient in the steam cooling period is between 20 and 120 W/m²K, which is dependent on the core steam flow rate mainly determined by the system pressure and total steam discharge area.

3) The convective heat transfer coefficient in the spray cooling period in the ROSA-III experiment is about 70 W/m²K. Relatively large standard deviation was observed in the mist cooling heat transfer coefficient data among the rods in the same bundle. The peak cladding temperature was observed at the heater rod which was not cooled well in the spray cooling period in each experiment.

4) Top-down quench has two-dimensional nature and becomes prominent with decreasing break area. All rods are not quenched from top of the core before a recovery of mixture level even in a small break test.

5) There is a different heat transfer behavior between small and large break experiments in a reflooding period. The heat transfer rate is the same as in the spray cooling period before a mixture level recovery and increases to a relatively high value after level recovery in a small break test. Whereas in a large break experiment, liquid entrainment is observed in core steam flow above the mixture level which enhances the heat transfer above a mixture level.

6) Measured bottom-up quench temperatures in the ROSA-III tests are expressed by a simple equation, a sum of saturation temperature and a constant of 262 K, with a standard deviation of 30 K.

7) The RELAP4/MOD6/U4/J3 code was revised to use the simple heat transfer model based upon the heat transfer coefficients and the bottom-up quench model obtained in the present heat transfer analysis results. The code can predict well the trend of the cladding surface temperature behavior.

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Fig. 17.1 Pressure transients
Fig. 17.2 Mixture level transients in the high power bundle
Fig. 17.3 Temperature transients of cladding surface at midplane in the high power bundle
Fig. 17.4 Details of heater rod
Fig. 17.5 Convective heat transfer coefficients versus time in the steam cooling period
Fig. 17.6 Convective heat transfer coefficients versus system pressure in the steam cooling period
Fig. 17.7 Convective heat transfer coefficients versus time in the spray cooling period
Fig. 17.8 Fraction of dried rods among thirty rods instrumented at midplane in the high power bundle versus dimensionless time (t − t_{dlyout}/(t_{qench} − t_{dlyout})
Fig. 17.9 Ratio of heat transfer rate to electrical power input rate versus time in the spray cooling period
Fig. 17.10 Total heat transfer coefficient versus time in the reflooding period
Fig. 17.11 Bottom-up quench superheat versus system pressure
Fig. 17.12 Comparison of rod surface temperature data in the high power channel and the analysis results using the original and improved versions of the RELAP4/MOD6/U4/J3 code
### Table 17.1 Experimental conditions

<table>
<thead>
<tr>
<th></th>
<th>Run 926</th>
<th>Run 916</th>
<th>Run 913</th>
<th>Run 912</th>
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<tr>
<td><strong>Break area %</strong></td>
<td>200</td>
<td>50</td>
<td>15</td>
<td>5</td>
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<td><strong>Measured initial conditions</strong></td>
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<tr>
<td>Steam dome pressure MPa</td>
<td>7.37</td>
<td>7.32</td>
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<tr>
<td>Lower plenum subcooling K</td>
<td>10.6</td>
<td>11.2</td>
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<td>Total jet pump discharge flow rate kg/s</td>
<td>16.3</td>
<td>16.5</td>
<td>16.4</td>
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<tr>
<td>Core outlet quality %</td>
<td>13.9</td>
<td>14.2</td>
<td>14.5</td>
<td>13.5</td>
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<td>Power kW</td>
<td>3967</td>
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### Table 17.2 Chronology of major events

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<td><strong>Break</strong></td>
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<tr>
<td>L2* level</td>
<td>3.0</td>
<td>6</td>
<td>13</td>
<td>19</td>
</tr>
<tr>
<td>L1* level</td>
<td>7.8</td>
<td>10</td>
<td>23</td>
<td>38</td>
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<tr>
<td>MSIV* closure</td>
<td>9</td>
<td>12</td>
<td>18</td>
<td>24</td>
</tr>
<tr>
<td>JPSU*</td>
<td>10</td>
<td>13</td>
<td>35</td>
<td>99</td>
</tr>
<tr>
<td>RLU*</td>
<td>13</td>
<td>18</td>
<td>48</td>
<td>150</td>
</tr>
<tr>
<td>LPF*</td>
<td>17</td>
<td>38</td>
<td>115</td>
<td>159</td>
</tr>
<tr>
<td>FWF*</td>
<td>68</td>
<td>142</td>
<td>250</td>
<td>323</td>
</tr>
<tr>
<td>LPCS actuation</td>
<td>71</td>
<td>143</td>
<td>248</td>
<td>318</td>
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<tr>
<td>LPCI actuation</td>
<td>96</td>
<td>183</td>
<td>326</td>
<td>406</td>
</tr>
<tr>
<td>ADS actuation</td>
<td>130</td>
<td>131</td>
<td>143</td>
<td>158</td>
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<td>Whole core uncover</td>
<td>72</td>
<td>105</td>
<td>198</td>
<td>275</td>
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<tr>
<td>Whole core recovery</td>
<td>137</td>
<td>210</td>
<td>366</td>
<td>440</td>
</tr>
</tbody>
</table>

*note - L2 level; 4.76 m from pressure vessel bottom
L1 level; 4.25 m from pressure vessel bottom
MSIV; main steam line isolation valve
JPSU; jet pump suction uncover
RLU; recirculation line nozzle uncover
LPF; lower plenum flashing
FWF; feedwater flashing

### Table 17.3 Recommended heat transfer data for calculation

<table>
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<th></th>
<th>Original</th>
<th>Revised</th>
<th></th>
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<tbody>
<tr>
<td>( h_{wet} )</td>
<td>1000 Btu/hr( \cdot F ) ( (5680 \text{ W/m}^2\text{K}) )</td>
<td>1000 Btu/hr( \cdot F ) ( (5680 \text{ W/m}^2\text{K}) )</td>
<td></td>
</tr>
<tr>
<td>( h_{dp} )</td>
<td>2.5 Btu/hr( \cdot F ) ( (14.2 \text{ W/m}^2\text{K}) )</td>
<td>12.3 Btu/hr( \cdot F ) ( (70 \text{ W/m}^2\text{K}) )</td>
<td></td>
</tr>
<tr>
<td>( h_{film} )</td>
<td>25 Btu/hr( \cdot F ) ( (142 \text{ W/m}^2\text{K}) )</td>
<td>44 Btu/hr( \cdot F ) ( (250 \text{ W/m}^2\text{K}) )</td>
<td></td>
</tr>
<tr>
<td>( X_{cri} )</td>
<td>0.05</td>
<td>0.05</td>
<td></td>
</tr>
<tr>
<td>( T_q )</td>
<td>-</td>
<td>( T_{sat} + 473 \text{ F} ) ( (262 \text{ K}) )</td>
<td></td>
</tr>
</tbody>
</table>
Fig. 17.1 Pressure transients.

Fig. 17.2 Mixture level transients in the high power bundle.
Fig. 17.3 Temperature transients of cladding surface at midplane in the high power bundle.

Fig. 17.4 Details of heater rod.
**Fig. 17.5** Convective heat transfer coefficients versus time in the steam cooling period.

**Fig. 17.6** Convective heat transfer coefficients versus system pressure in the steam cooling period.
Fig. 17.7 Convective heat transfer coefficients versus time in the spray cooling period.

Fig. 17.8 Fraction of dried rods among thirty rods instrumented at midplane in the high power bundle versus dimensionless time \( \frac{t - t_{\text{dryout}}}{t_{\text{quench}} - t_{\text{dryout}}} \).
Fig. 17.9  Ratio of heat transfer rate to electrical power input rate versus time in the spray cooling period.

Fig. 17.10  Total heat transfer coefficient versus time in the reflooding period.
Fig. 17.11  Bottom-up quench superheat versus system pressure.

Fig. 17.12  Comparison of rod surface temperature data in the high power channel and the analysis results using the original and improved versions of the RELAP4/MOD6/U4/J3 code.
18. BWR Large Break LOCA Counterpart Tests at ROSA-III and FIST

18.1 Introduction

The General Electric Company (GE) performed various safety related transient tests in the Full Integral Simulation Test (FIST) facility\(^1\),\(^2\), the flow diagram of which is shown in Fig. 18.1. The FIST facility, scaled to a 624 bundle BWR/6-218\(^2\), has one full length electrically heated bundle.

The first objective of this chapter is to develop common understanding and interpretation of the controlling phenomena observed in large break LOCA tests performed in the ROSA-III and FIST facilities\(^3\). Large break tests in a recirculation pump suction line (ROSA-III Run 983\(^4\) and FIST Run 6DBA1B\(^5\)) were chosen for the counter part test study\(^6\). Each test was performed under similar test conditions in each facility. Evaluation of the test comparisons consists of three parts. First is the similarity of the test conditions, i.e. the initial conditions and boundary conditions such as the power, ECC flow and break flow. Second is the similarity of system responses to the test conditions, such as timing of key events, system pressure, liquid level and fuel rod surface temperature transients. Third is the effect of facility differences and different scaling compromises in the two facilities.

The second objective of this study is to examine the similarity of the thermal-hydraulic phenomena between ROSA-III and FIST large break tests and a BWR large break LOCA. For this similarity study, at first, ROSA-III Run 983 and FIST Run 6DBA1B were analyzed by the RELAP5/Mod1 (Cycle 018) code\(^7\) to examine capability of the code to calculate large break LOCAs. Then, a BWR counterpart was analyzed by using almost the same analysis methodology as was used in the ROSA-III and FIST analyses. Finally, the similarity between ROSA-III and FIST large break tests and a BWR large break LOCA was studied by comparing calculated results for the ROSA-III and FIST tests and a BWR LOCA. The primary concern of the present analyses is to compare the fundamental thermal-hydraulic phenomena, such as system pressure and fuel rod surface temperature transients.

18.2 Comparison of Test Facilities

The objective for both ROSA-III and FIST facilities is for the scaled test apparatus to simulate, on a real time basis, BWR system thermal-hydraulic response following a postulated LOCA. Both facilities have the capability to establish initial thermal-hydraulic conditions typical of a BWR, as well as the important BWR system features that govern the mass and energy transfer rates.

The ROSA-III facility is volumetrically scaled (1/424) to a BWR/6-251 with 848 fuel bundles. The four half length bundle concept was adopted to study thermal-hydraulic interaction among the bundles. The FIST facility is volumetrically scaled (1/624) to a BWR/6-218 with 624 fuel bundles. The full height vessel and single electrically heated bundle provide full scale values for fluid conditions, velocities, and static heads. The vessel regions are shown in Fig. 18.2.

An overall comparison of the ROSA-III and FIST facilities is given in Table 18.1. This table summarizes the pertinent information on the bundles, region volumes and heights, and the break areas.
The bundles in each facility consists of 62 heated rods and two water rods in an 8 x 8 square array. The rods are electrically heated with a chopped cosine axial power distribution and an axial peaking factor of 1.4, as shown in Fig. 18.3. The 1.88 m ROSA-III bundles and the FIST 3.81 m bundle have equivalent axial power shapes when compared on a normalized length basis as seen in Fig. 18.3. Also shown in this figure are the relative locations of the rod thermocouples in each bundle. The ROSA-III bundles have a local peaking factor of 1.1 and FIST 1.042, but this difference has a negligible effect. ROSA-III simulation of two full size bundles with four half length bundles doubles the core flow area. The flow areas of the four inlet orifices and upper tieplates are reduced to preserve the core region CCFL performance characteristics, as seen in Table 18.1. The relative heights and elevations of the bundles are shown in Fig. 18.4.

The volume of each test vessel region is scaled to the corresponding volume in the reference BWR. The regional volumes are compared in Table 18.1 on a per-full-size-bundle basis (on a basis of one bundle with full size), since ROSA-III has four half length bundles and FIST is scaled to one. The scaled system volumes are within six percent. The ROSA-III upper plenum and recirculation loop regions are larger than FIST, and the other regions are correspondingly smaller.

The total water volumes, however, which are given in Table 18.2, compare within ten percent. The distribution of the downcomer fluid volumes with respect to the Level 1 and Level 2 trip elevations is important because ECCS flow, MSIV closure, and ADS activation are initiated by these settings. These elevations and volumes are given in Table 18.3.

Height scaling is also compared in Table 18.1 and Fig. 18.4. The FIST facility, with full scale elevations, preserves region cross-sectional areas, region void fractions, and elevation heads. The shorter scaling in ROSA-III does not have a large effect on overall system response. The simulated breaks are placed in the same relative locations so that the mass and energy fluxes are simulated, and are sized to give mass and energy flow rates in proportion to the volume scaling. The large break areas (the break areas in the large break tests) are only six percent different between the two facilities. Because they have different reference systems, the FIST recirculation suction line large break area is 4% larger than ROSA-III, and the drive line large break is 22% larger. The small break areas are within 2%. The FIST steamline break area is 25% larger than ROSA-III for the first 5.5 seconds of the transient, and equal thereafter. Small break and steamline break counterpart tests will be described in Chapters 19 and 20, respectively.

The stored heat in the metal structures in the regions containing water has the greatest impact on the depressurization rate. In these regions, which include the lower plenum, guide tube, bypass, bundles, and upper plenum, ROSA-III has 36% more metal per full size bundle than FIST. Localized stored heat effects are also significant. For example, in FIST the large bypass metal mass caused refilling oscillations for some tests. The large lower plenum metal mass in ROSA-III increases the flashing in this region.

Overall the match between ROSA-III and FIST is quite good. The major differences are the number of core flow paths (parallel channel flow) and the region heights.

18.3 Test Conditions

Comparison of initial conditions is made in Table 18.4 for ROSA-III Run 983 and FIST Run 6DBA1B. The LPCI-diesel generator (DG) is assumed to fail in both tests, resulting in the failure of two out of three LPCI systems. The initial system pressures are the same. Since two full bundles in ROSA-III are equivalent to one in FIST, the initial power and flow rates
in ROSA-III should be twice those in FIST, i.e., on a per-full-size-bundle basis the initial powers and flows should be equal. However, the ROSA-III initial power is 64% lower than in FIST due to power supply limitation. Therefore, the recirculation line flow rate is controlled to give the desired lower plenum subcooling, which is seen to be very nearly the same in both facilities. Steam flow and feedwater flow in ROSA-III are lower, corresponding to the lower initial core power. The downcomer levels, different in the two facilities because of different height scaling, are set to give the correct scaled water volume in the downcomer. The lower initial power reduces the bundle and upper plenum void fractions, and therefore increases the initial mass in these regions.

The boundary conditions are also compared in Table 18.4. The ROSA-III bundle power, which is initially low due to power supply limitation, is held constant until the bundle power is very nearly equal to the reference bundle power, i.e. 9 s after break. From 9 s to 27 s after the break the ROSA-III power curve is slightly higher than FIST, and thereafter the powers are equal. In both facilities, the break initiation is defined as the beginning of the test, i.e. time zero. The recirculation suction break areas, which are the dominant breaks in these tests, are very nearly equal. The drive line break area in ROSA-III is 28% smaller than in FIST. The sum of the two break areas is 6.5% smaller in ROSA-III. There is no activation of the ADS in FIST, whereas in ROSA-III the ADS is actuated at 115 s. This does not have a significant effect because the system pressure is low by this time. The recirculation pumps are tripped off at time zero in both tests. The feedwater valve is closed at time zero in FIST and at 2 to 4 s in ROSA-III. In FIST, the pressure control system is actuated to simulate the BWR pressure control system which maintains the turbine inlet pressure at 6.7 MPa by regulating the turbine valve. In ROSA-III, the pressure control system is operated to maintain the steam dome pressure at 6.7 MPa. In ROSA-III the MSIV closure is initiated at 13 s after the break. In FIST the MSIV closure is activated by the Level 1 trip with a 2 s delay. All the ECC systems are activated by time trips in both tests, the HPCS at 27 s in each and the LPCS/LPCI at 50 s/50 s and 35 s/35 s in ROSA-III and FIST, respectively. Injections of LPCS and LPCI are also specified to activate at system pressures below 1.78 MPa and 1.55 MPa, respectively, in ROSA-III, and below 2.00 MPa and 1.74 MPa, respectively, in FIST. In FIST the broken recirculation loop is isolated at time zero and the intact loop at 13 s, to prevent flashing from these overscale volumes. The loops are not isolated in ROSA-III.

18.4 Comparison of Test Results

18.4.1 System Pressure and Sequence of Events

Timing of the key events that follow is compared in Table 18.5 and the system pressures are compared in Fig. 18.5.

In the ROSA-III test, the system pressure decreased after break due to fluid discharge through the break. The system pressure was maintained at 6.7 MPa from 6 s to 13 s after the break due to the actuation of the pressure control system. The pressure started to increase after the closure of the main steam isolation valve (MSIV) from 13 s to 15 s after break. The mixture level in the downcomer decreased rapidly after break and the recirculation line outlet in the downcomer was uncovered (RLU) at 17 s after break. Then the steam in the vessel was discharged directly through the recirculation suction break and the system pressure began to decrease rapidly. At 18 s after break, the system pressure fell below 6.4 MPa, and the lower plenum fluid reached the saturation condition and began to flash. The depressurization rate slowed down after the initiation of the lower plenum flashing (LPF) because of continuous steam generation in the lower plenum. The HPCS was actuated at
27 s after break and the LPCS and LPCI were actuated at system pressures of 1.78 MPa and 1.55 MPa, respectively, in accordance with the ECCS actuation conditions.

The difference in the system pressure transient immediately after the break initiation is due to differences in the pressure control system operation logic, core power curves during first 9 s and recirculation suction break uncover timing. The ROSA-III MSIV closed approximately 3 s before the recirculation suction break was uncovered, causing a slight increase in the pressure just before the rapid depressurization began. The FIST MSIV closure occurred as the break was uncovered and the impact was not noticeable. The rapid depressurization due to recirculation suction break uncover began at 8 s in FIST and at 17 s in ROSA-III. As a consequence, the FIST system pressure remained slightly lower than ROSA-III throughout the transient. After recirculation suction break uncovery, the depressurization rates in both tests were almost equal. On the whole, the system pressure transients in the ROSA-III Run 983 and FIST Run 6DBA1B almost agreed with each other.

18.4.2 Mixture Level

The mixture level transients in the downcomer measured with conductivity probes are shown in Fig. 18.6 for the ROSA-III and FIST tests. The mixture level in the downcomer was normalized by the length between the jet pump suction nozzle elevation and the recirculation line suction nozzle elevation in both tests.

In the ROSA-III test, the mixture level in the downcomer fell rapidly after the break due to the loss of primary coolant through the break. The jet pump suction were uncovered at 12 s after break and the exit nozzles of the recirculation line at 17 s. The mixture level in the downcomer was recovered considerably later than that in the core, since ECCS water was injected inside the core shroud.

The downcomer mixture level in the FIST test fell slightly earlier than that in ROSA-III test. The jet pump suction in the FIST were uncovered at 5 s after the break and the exit nozzles of the recirculation line at 8 s. Recovery of the downcomer mixture level in the FIST test occurred somewhat later than in ROSA-III test. On the whole, however, the downcomer mixture level transients in ROSA-III Run 983 and FIST Run 6DBA1B almost agreed with each other.

The mixture level transients inside the core shroud are presented in Fig. 18.7 for the ROSA-III and FIST tests. The mixture level in the ROSA-III test was determined from measured conductivity probe signals, while the mixture level in the FIST Test was estimated from nodal densities which were calculated from measured nodal differential pressures. The mixture level in the core for ROSA-III test presented in Fig. 18.7 is for the B bundle core. In both tests, the mixture levels in the upper plenum, the core and the lower plenum were normalized by the lengths between the top of upper plenum and the upper tie plate, between the upper tie plate and the lower tie plate, and between the core inlet orifice and the inlet from jet pump, respectively.

In the ROSA-III test, the mixture level was formed in the lower plenum at 27 s following the initiation of the lower plenum flashing, though almost no core was uncovered at this time. This indicates the occurrence of counter current flow limiting (CCFL) at the core inlet orifice. As the flashing in the lower plenum subsided, the mixture level was formed at 35 s after the break in the bundle B core, and the level decreased from Position 1 to Position 3 between 35 s and 72 s. The mixture level in the bundle B core began to recover from approximately 90 s and the whole bundle B core was reflooded at 115 s following actuations of LPCS and LPCI from 85 s and 95 s, respectively. In the ROSA-III test, while uncovery of the bundle B and D cores was observed, the bundle A and C cores were being covered with two-phase
mixture throughout the test. Measured core inlet flow rates showed that the bundles A and C were upflow while the bundles B and D in downflow when the mixture level was being formed in the lower plenum. The mixture level in the lower plenum started to rise at approximately 135 s due to actuations of LPCS and LPCI at 85 s and 95 s, respectively. Judging from conductivity probe signals in the upper plenum, it is considered that the upper plenum was being covered with two-phase mixture throughout the test.

In both tests, soon after the lower plenum flashing, the mixture level was formed in the lower plenum. This occurred at 18 s in FIST and at 27 s in ROSA-III. As a consequence, until the lower plenum was refilled, the water fall back from the bundle was limited by CCFL at the core inlet orifice. However, in FIST test, after the liquid level in the lower plenum decreased to the inlets from the jet pumps at 39 s and the lower plenum steam was vented up the jet pumps, the water fall back from the bundle and core uncoverly increased. A rapid decrease in the core mixture level after 32 s in FIST test is probably due to this increased water fall back arising from CCFL break-down at the core inlet orifice. On the other hand, sudden recovery of the core mixture level at 125 s after the break in FIST test is due to water fall back from the upper plenum caused by CCFL break-down at the upper tie plate. The multi-channel effect on core mixture level transient was observed in the ROSA-III test, as was stated in the above paragraph. However, the trends of the mixture level transients inside the core shroud in ROSA-III Run 983 and FIST Run 6DBA1B agreed well with each other.

18.4.3 Surface Temperature of Fuel Rod

The surface temperatures of fuel rods in the ROSA-III and FIST tests are compared in Fig. 18.8. The surface temperatures were measured at several elevations of the core in both tests, as shown in the figure. While the surface temperatures presented in the figure for ROSA-III test are those for the fuel rod D-22, the surface temperatures for FIST test are those obtained from several rods, as shown in the figure. The location of the rod indicated in Fig. 18.8 is shown in Fig. 18.9. Temporary dryout of the fuel rod surface before initiation of the lower plenum flashing due to occurrence of departure from nucleate boiling (DNB) was not observed in both tests.

In the ROSA-III test, after the lower plenum flashing became less effective, the fuel rod surface temperatures started to rise successively from Position 1 to Position 3 between 39 s and 48 s after the break. Generally, there is a strong correlation between the fuel rod surface temperature and the mixture level transient in the core and the initiation of fuel rod surface temperature at a given location corresponds to the uncoverly time of the fuel rod surface at the same location due to the fall in the core mixture level. However, core mixture level signals in the bundle D were not recorded in the test, while the core mixture level in the bundle B was measured as presented in Fig. 18.7. The fuel rod surface at Position 3 repeated dryout and rewet from 48 s to 93 s probably due to the oscillation of the mixture level around this position. After the initiation of LPCS and LPCI at 85 s and 95 s, respectively, the core was quenched upward from Position 3 to Position 1 between 93 s and 95 s. The peak cladding temperature (PCT) in ROSA-III test was 575 K observed at Position 2.

Core heat-up began at very nearly the same time in both the tests, i.e. at 39 s in ROSA-III and 41 s in FIST. In FIST there was drainage through the upper tie plate that cooled the very top of the bundle, Positions 1 through 3. The ROSA-III bundle D, which showed downflow at the core inlet orifice, was heated up at the very top of the bundle, Positions 1 through 3, indicating liquid loss to the lower plenum and little drainage from the upper plenum. The ROSA-III bundle B was also heated up slightly later. However, the ROSA-III bundles A and
C were not heated up at all. These results are consistent with the mixture level transients or the core inlet flow directions in the four bundles mentioned earlier. The HPCS, LPCS and LPCI were actuated at 27, 68 and 75 s, respectively, in FIST test and increased upper tieplate drainage cooled the top of the fuel rods, limiting the PCT to 656 K at 110 s. Counter-current flow limiting break-down at 125 s was indicated by the sudden slight drop in temperature at the upper elevation in Fig. 18.8. In ROSA-III test, the HPCS was also actuated at 27 s, and the LPCS and LPCI at 85 s and 95 s, respectively. Subcooled water was present at the peak power position, Position 4, in the bundle D at 160 s. The hot bundles (the bundles which showed heat-up) in both tests were in counter-current flow with some cooling from the top. The PCT was 81 K higher in FIST since it occurred at a higher heat flux location.

On the whole, however, the trends of the fuel rod surface temperatures in the hot bundle in ROSA-III Run 983 and FIST Run 6DBA1B agreed with each other.

18.5 Analysis

Post-test analyses of ROSA-III Run 983 and FIST Run 6DBA1B were performed with the RELAP5/Mod1 (cycle 018) code. The objective of these analyses was to examine the similarity between the thermal-hydraulic phenomena in ROSA-III and FIST large break tests and a BWR large break LOCA. The primary objective of these analyses was to compare the fundamental trends in the transients of system pressure and fuel rod surface temperatures. For the study of similarity, ROSA-III Run 983 and FIST Run 6DBA1B were first analyzed with the code to examine the capability of the code to calculate the transients. Then, a BWR counterpart was analyzed by using almost the same analytical methodology as was used in the ROSA-III and FIST analyses. Finally, the similarity between ROSA-III and FIST large break tests and a BWR large break LOCA was examined by comparing the calculated results for the ROSA-III and FIST tests and the BWR LOCA.

18.5.1 ROSA-III Analysis

The ROSA-III system was modeled with 66 volumes, 74 junctions and 26 heat structures as shown in Fig. 18.10. The core was divided into two regions: one modeled one out of the four bundles and the other modeled the other three bundles. Each region had 7 heat structures which modeled fuel rods and corresponded to the 7 steps of the axial power distribution. Since no jet pump model was incorporated in the code, small pumps with the same coastdown and homologous curves as the recirculation pumps were added to the suction lines of the jet pumps to establish the steady-state flow condition in the system. Measured flow rates were used as a function of system pressure in the analysis for the main steam flow rate, ADS flow rate, feedwater flow rate and ECCS flow rates. Closure of MSIV, opening of ADS valve, closure of feedwater line valve and actuations of ECCS were done at the same specified times or pressures as in the test. The pressure control system was modeled to keep the system pressure above 6.7 MPa before MSIV closure as was done in the test. The discharge coefficient was set to 0.6 in the calculation of break flow rate.

The calculated system pressure is compared with the measured result in Fig. 18.11. The calculated system pressure agrees well with the measured pressure during rapid decrease after break, hold by the pressure control system, recovery after MSIV closure and rapid drop after uncovering of the recirculation line outlet in the downcomer. The time of uncovering of the recirculation line outlet in the downcomer is a little earlier in the analysis than in the test. The depressurization rate slows down somewhat after the initiation of lower plenum flashing in both calculated and measured results. The overall agreement in the behavior of
the calculated and measured system pressure transients is satisfactory.

The calculated surface temperatures of fuel rods are compared with the measured results in Fig. 18.12. The measured results are surface temperatures for the fuel rod D-22, the same as those presented in Fig. 18.8. Temporary dryout immediately after the break is calculated at Positions 1 and 2 in the analysis, while it was not observed in the test. This is due to the occurrence of departure from nucleate boiling (DNB) arising from decrease in the core inlet flow rate in the analysis. The core inlet flow rate immediately after the break is closely related with the characteristics of jet pumps and coast down of main recirculation pumps. Hence, it is necessary to reexamine DNB correlations or a jet pump model used in the code, or input for main recirculation pumps used in the analysis. The fuel rod surface temperatures at Positions 1 and 2 are rewetted immediately after the lower plenum flashing in the analysis. The time of dryout after mitigation of the lower plenum flashing and the increasing rate of surface temperature in the upper part of the core in the analysis agree well with those in the test. However, the dryout after mitigation of the lower plenum flashing is calculated excessively in the middle and lower part of the core in the analysis, while almost no dryout was observed in the test. This is because the mixture level in the core decreases to the lower tie plate in the analysis due to insufficient calculation of CCFL at the core inlet orifice. Hence, it is necessary to reconsider interphase drag correlations used at flow contractions. On the whole, the surface temperature of fuel rod is calculated well in the upper part of the core and is estimated conservatively in the middle and lower part of the core in the analysis.

18.5.2 FIST Analysis

The FIST test was also analyzed with the RELAP5/MOD1 (cycle 018) code using almost the same methodology as was used in the ROSA-III analysis. The FIST system was modeled with 58 volumes, 65 junctions and 27 heat structures. The core consisted of one region which modeled the FIST bundle. The region had 5 heat structures which modeled fuel rods. Measured jet pump discharge flow rates were injected into the lower plenum as a function of time in the analysis, since enough informations on the characteristics of jet pumps and coast down of main recirculation pumps were not available. The pressure control system was modeled to keep the system pressure above 7.2 MPa before MSIV closure.

The calculated system pressure is compared with the measured result in Fig. 18.13. The calculated system pressure reproduces well the rapid decrease after break, the hold by the pressure control system and the rapid drop after uncover of the recirculation line outlet in the downcomer. The recirculation line outlet in the downcomer is uncovered a little earlier in the analysis than in the test as was the case with ROSA-III. The depressurization rate slows down somewhat after the initiation of lower plenum flashing in both calculated and measured results also as with the ROSA-III. The overall agreement in the behavior of the calculated and measured system pressure transients is satisfactory.

The calculated surface temperatures of fuel rods are compared with the measured results in Fig. 18.14. The measured results are the same surface temperatures as presented in Fig. 18.8. As is the case with ROSA-III, temporary dryout immediately after the break is calculated at Positions 3 and 4 in the analysis, while it did not occur in the test. This is also due to the occurrence of DNB in the analysis. Hence, reexamination of DNB correlations in the code is required. The behavior of dryout and quench of fuel rod after mitigation of the lower plenum flashing in the upper and middle part of the core in the analysis agrees well with the behavior in the test. Especially, the trend of top-down quench in the upper and middle part of the core is predicted well in the analysis. However, the dryout after mitigation of the lower plenum flashing is overestimated in the lower part of the core in the analysis, while
almost no dryout occurred in the test. The reason is the same as that described for the ROSA-III case. On the whole, the surface temperature of fuel rod is predicted well in the upper and middle part of the core, while it is estimated conservatively in the lower part of the core.

The ROSA-III Run 983 and FIST Run 6DBA1B were analyzed with the RELAP5/MOD1 (cycle 018) code to examine the capability of the code to calculate large break LOCAs. The code reproduced the system pressures in both the tests considerably well. The code also calculated relatively well the surface temperatures of fuel rod in both the tests, while the surface temperatures in the lower part of the core were estimated conservatively.

18.5.3 BWR Analysis and Similarity between ROSA-III and FIST Tests and BWR LOCA

The BWR counterpart was analyzed with the RELAP5/MOD1 (cycle 018) code using almost the same methodology as was used in the ROSA-III and FIST analyses. The BWR system was modeled by 67 volumes, 75 junctions and 31 heat structures. The core was divided into two regions: one modeled the central core with 748 bundles and the other modeled the peripheral core with 100 bundles. The central core included 7 heat structures divided along axial direction with radial and local peaking factors of 1.4 and 1.13, respectively, and 7 heat structures with radial and local peaking factors of 1.0 and 1.00, respectively. The peripheral core had 7 heat structures with radial and local peaking factors of 0.7 and 1.00, respectively. The axial peaking factors of these three heat structure groups were 1.4, as were the cases with ROSA-III and FIST analyses. The power curve used in the analysis was that evaluated by General Electric Company including 3σ uncertainty\(^9\). The main steam flow rate and the feedwater flow rate were kept at constant values until these flows were tripped. The HPCS, LPCS and LPCI flow rates were given as a function of the system pressure. The feedwater supply was stopped at 4 s and the MSIV was closed by the L2 level signal with a time delay of 3 s\(^10\). The ADS and HPCS were actuated by the L1 level signal with a time delay of 120 s and the L2 signal with a delay of 27 s, respectively, and the LPCS and LPCI were actuated at system pressures of 2.2 MPa and 1.6 MPa, respectively\(^10\). The pressure control system was modeled to keep the system pressure above 6.7 MPa before MSIV closure as was the case with ROSA-III analysis.

The calculated system pressure transient for a BWR is compared with the calculated results for the ROSA-III and FIST in Fig. 18.15. The system pressure transient until the uncovering of recirculation line outlet in the BWR analysis agrees well with that in the ROSA-III analysis. This is because the pressure control system in the BWR analysis was modeled in the same way as used in the ROSA-III analysis. The time of the uncovering of recirculation line outlet in the BWR analysis agrees with that in the ROSA-III analysis better than that in the FIST analysis. This means that the liquid level transient in the downcomer in a BWR is closer to that in the ROSA-III than that in the FIST. The decrease in depressurization rate due to the lower plenum flashing occurred at 6.4 MPa is seen in all the three analyses. The temporary hold of system pressure after LPCS actuation is seen little in the BWR analyses, while it was calculated in both the ROSA-III and FIST analyses. The depressurization rate after the uncovering of recirculation line outlet in the BWR analyses is somewhat larger than those in the ROSA-III and FIST analyses. The reason may be that the stored heat in the vessel walls and vessel internals slows down the depressurization rate in the ROSA-III and FIST. On the whole, however, the overall trend of the system pressure transient in the BWR analysis agrees well with those in the ROSA-III and FIST analyses.

The calculated surface temperatures of fuel rods with radial and local peaking factors of 1.0 and 1.00, respectively, in the central core in the BWR analysis are compared with corresponding surface temperatures in the ROSA-III and FIST analyses in Fig. 18.16. The
temporary dryout immediately after the break due to occurrence of DNB is seen at Positions 1 through 4 in the BWR analysis. This agrees well with the results in the FIST analysis. In the ROSA-III analysis, the temporary dryout immediately after the break is calculated only at Positions 1 and 2. The main reason may be that the ROSA-III core power was held constant, at 40% of scaled steady state power, during first 9 s. However, it should be noted that no dryout was observed immediately after the break in both the ROSA-III and FIST tests. The times of dryouts after mitigation of the lower plenum flashing in the BWR analysis agree with those in the FIST analysis better than those in the ROSA-III analysis, especially in the middle part of the core. On the other hand, the quench behavior in the BWR analysis agrees with that in the ROSA-III analysis better than that in the FIST analysis, especially in the middle and lower part of the core. The fuel rod surface temperatures in the upper part of the core repeat rewet and dryout after LPCS actuation in the BWR analysis. This corresponds to intermediate behavior between bottom-up quench in the ROSA-III analysis and top-down quench in the FIST analysis. On the whole, however, the overall trends of the fuel rod surface temperature transients in the BWR analysis agree well with those in the ROSA-III and FIST analyses.

From the above discussions, it has become clear that the fundamental phenomena during a BWR large break LOCA are simulated well by ROSA-III and FIST large break tests. Therefore, ROSA-III and FIST large break test data are useful to study the fundamental phenomena of a BWR large break LOCA and to assess a computer code for a BWR large break LOCA.

18.6 Conclusions

A large break test in a recirculation pump suction line was conducted at the ROSA-III test facility in JAERI and at the FIST test facility in GE Company under the same test conditions as far as possible in order to develop common understanding and interpretation of the controlling thermal-hydraulic phenomena during a large break LOCA of a BWR. The LPCI-DG was assumed to fail in both tests. The following conclusions were obtained from the comparison of results of both tests.

(a) The agreement between ROSA-III and FIST system responses is very good. In particular the system pressure responses agree well.

(b) The mixture level and fuel rod surface temperature transients in the ROSA-III hot bundles are similar to those of the FIST bundle.

(c) The half height scaling in ROSA-III does not have a significant influence on the system response.

(d) A difference in system response is a little faster system depressurization in FIST caused by early recirculation suction break uncovering attributed to facility differences. The faster system depressurization in FIST also results in earlier actuations of LPCS and LPCI by about 20 s.

(b) Larger bundle uncoverage occurs in FIST than in ROSA-III. Lower plenum steam in the FIST flows out through jet pumps when they are uncovered allowing more liquid to drain from the bundle.

The ROSA-III and FIST tests and a corresponding BWR LOCA were analyzed with the RELAP5/MOD1 (cycle 018) code in order to study the similarity of ROSA-III and FIST large break tests to a BWR large break LOCA. The following conclusions have been obtained from the comparison of calculated results.

(1) The RELAP5/MOD1 (cycle 018) code reproduces both test results relatively well. This indicates that the code has the capability to calculate relatively well the thermal-hydraulic
behavior during a large break LOCA of a BWR. However, it is desirable to reexamine DNB correlations and interphase drag correlations used in the code at flow contractions. (2) The fundamental thermal-hydraulic phenomena, such as the system pressure and the fuel rod surface temperatures, during the ROSA-III and FIST large break tests are similar to those during a BWR large break LOCA, though there are some differences in details.

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### Table 18.1 Comparison of ROSA-III and FIST facilities

<table>
<thead>
<tr>
<th>Item</th>
<th>ROSA-III Total</th>
<th>Per Full Size Bundle</th>
<th>FIST Total</th>
<th>Per Full Size Bundle</th>
<th>ROSA-III/FIST Total</th>
<th>Per Full Size Bundle</th>
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<tr>
<td>Bundles</td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
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<td>8X8</td>
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<tr>
<td>No. of Bundles</td>
<td>41</td>
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<td>1</td>
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<td>Heated Length (m)</td>
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<td></td>
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<td>Local Peaking Factor</td>
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<td>1.04</td>
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<td>Axial Peaking Factor</td>
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<td></td>
<td>1.40</td>
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<td>No. of Heated Rods/Bundle</td>
<td>62</td>
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<td>62</td>
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</tr>
<tr>
<td>No. of Water Rods/Bundle</td>
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<td></td>
<td>2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>O.D. of Heated Rods (mm)</td>
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<td></td>
<td>12.27</td>
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<td>Flow Area (cm²)</td>
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<td>96.8(1)</td>
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<td>UTP Flow Area (cm²)</td>
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<td>59.5(1)</td>
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<td>SEO Flow Area (cm²)</td>
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<td>30.4(1)</td>
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<tr>
<td>Maximum Power (MW)</td>
<td>4.46(2)</td>
<td>2.23</td>
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<td></td>
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<td>Volumes (m³)</td>
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<tr>
<td>Total System</td>
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<td>0.709</td>
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</tr>
<tr>
<td>Steam Dome</td>
<td>0.371(5)</td>
<td>0.186</td>
<td>0.218</td>
<td>0.85</td>
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</tr>
<tr>
<td>Downcomer</td>
<td>0.340(5)</td>
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<td>0.170</td>
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<td></td>
</tr>
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<td>Jet Pumps ; Recirc. Loops</td>
<td>0.172</td>
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<td>0.024(3)</td>
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<td>Steam Separator</td>
<td>0.031</td>
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</tr>
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<td>Upper Plenum</td>
<td>0.124</td>
<td>0.062</td>
<td>0.044</td>
<td>1.41</td>
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</tr>
<tr>
<td>Bundles</td>
<td>0.096</td>
<td>0.048</td>
<td>0.043</td>
<td>1.12</td>
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<td></td>
</tr>
<tr>
<td>Bypass</td>
<td>0.060</td>
<td>0.030</td>
<td>0.037</td>
<td>0.81</td>
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</tr>
<tr>
<td>Lower Plenum</td>
<td>0.167</td>
<td>0.083</td>
<td>0.088</td>
<td>0.94</td>
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</tr>
<tr>
<td>Guide Tubes</td>
<td>0.057</td>
<td>0.028</td>
<td>0.042</td>
<td>0.67</td>
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<td>Heights (m)</td>
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<tr>
<td>Total Vessel</td>
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<td>19.42</td>
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<tr>
<td>Steam Dome</td>
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<td>6.09</td>
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<td>Downcomer</td>
<td>4.51</td>
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<td>10.80</td>
<td>0.42</td>
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<tr>
<td>Jet Pumps</td>
<td>2.41</td>
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<td>4.50</td>
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<tr>
<td>Steam Separator</td>
<td>1.12</td>
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<td>2.10</td>
<td>0.53</td>
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</tr>
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<td>Upper Plenum</td>
<td>0.69</td>
<td></td>
<td>1.83</td>
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<td>Bypass</td>
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<td>4.39</td>
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<td>Lower Plenum</td>
<td>1.28</td>
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<td>4.11</td>
<td>0.31</td>
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<tr>
<td>Break Flow Areas (mm²)</td>
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<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ADS</td>
<td>349.6(6)</td>
<td>174.8</td>
<td>186.7</td>
<td>0.94</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Large Recirc. Break</td>
<td>619.2</td>
<td>309.6</td>
<td>331.4</td>
<td>0.94</td>
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<td></td>
</tr>
<tr>
<td>Recirc. Suction</td>
<td>539.1</td>
<td>269.5</td>
<td>279.6</td>
<td>0.96</td>
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<td></td>
</tr>
<tr>
<td>Drive Line</td>
<td>80.1</td>
<td>40.0</td>
<td>51.8</td>
<td>0.78</td>
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<tr>
<td>Small Recirc. Break</td>
<td>14.52</td>
<td>7.26</td>
<td>7.44</td>
<td>0.98</td>
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<tr>
<td>Steamline Break</td>
<td>754.8</td>
<td>377.4</td>
<td>501/377(4)</td>
<td>0.75/1.00</td>
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</tr>
</tbody>
</table>

(1) Four, half-length bundles were used in ROSA-III.
(2) Power decay delayed 9 seconds to compensate for low initial power.
(3) After loop isolation.
(4) FIST streamline break area changes from 501 to 377 mm² at 5.5 seconds.
(5) Based on a water level of 5.0 m which is used in normal ROSA-III tests.
(6) Normal ADS area for ROSA-III tests is 188.6 mm².
### Table 18.2  Fluid and water volumes

#### SCALING OF FLUID VOLUMES

<table>
<thead>
<tr>
<th>Volumes</th>
<th>$\frac{1}{2} \times$ ROSA-III (m$^3$)</th>
<th>FIST (m$^3$)</th>
<th>$\frac{1}{2}$ ROSA-III/FIST</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total Vessel</td>
<td>0.709</td>
<td>0.438</td>
<td>0.712</td>
</tr>
<tr>
<td>Steam Dome</td>
<td>0.186</td>
<td></td>
<td>0.218</td>
</tr>
<tr>
<td>Downcomer</td>
<td>0.170(5)</td>
<td>0.170(6)</td>
<td>0.170</td>
</tr>
<tr>
<td>Jet Pumps/Loops</td>
<td>0.086</td>
<td>0.086</td>
<td>0.024(1)</td>
</tr>
<tr>
<td>Steam Separators</td>
<td>0.016</td>
<td>0.004(2)</td>
<td>0.047</td>
</tr>
<tr>
<td>Upper Plenum</td>
<td>0.062</td>
<td>0.017(2)</td>
<td>0.044</td>
</tr>
<tr>
<td>Bundles</td>
<td>0.048</td>
<td>0.020(6)</td>
<td>0.043</td>
</tr>
<tr>
<td>Bypass</td>
<td>0.030</td>
<td>0.030</td>
<td>0.037</td>
</tr>
<tr>
<td>Lower Plenum</td>
<td>0.083</td>
<td>0.083</td>
<td>0.088</td>
</tr>
<tr>
<td>Guide Tubes</td>
<td>0.028</td>
<td>0.028</td>
<td>0.042</td>
</tr>
</tbody>
</table>

(1) After loops isolated.
(2) $\alpha = 0.73$, $x = 0.125$, homogeneous equilibrium, Run 952 initial conditions.
(3) $\alpha = 0.78$, $x = 0.16$, homogeneous equilibrium, Run 952 initial conditions.
(4) $\alpha = 0.59$, $x = 0.07$, homogeneous equilibrium, Run 952 initial conditions.
(5) $\alpha = 0.65$, $x = 0.09$, homogeneous equilibrium, Run 952 initial conditions.
(6) Based on water level of 5.0 m.

#### SUMMARY OF WATER VOLUMES

<table>
<thead>
<tr>
<th>Volumes</th>
<th>$\frac{1}{2}$ ROSA-III (m$^3$)</th>
<th>FIST (m$^3$)</th>
<th>Difference (m$^3$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Downcomer + Jet Pump/Loops</td>
<td>0.256</td>
<td>0.194</td>
<td>0.062</td>
</tr>
<tr>
<td>Separator + Upper Plenum + Bundles + Bypass</td>
<td>0.071</td>
<td>0.072</td>
<td>-0.001</td>
</tr>
<tr>
<td>Lower Plenum + Guide Tubes</td>
<td>0.111</td>
<td>0.130</td>
<td>-0.019</td>
</tr>
<tr>
<td>Total</td>
<td>0.438</td>
<td>0.396</td>
<td>0.042</td>
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</tbody>
</table>
### Table 18.3 Downcomer volumes

#### A. Elevations (m)

<table>
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<tr>
<th>Level</th>
<th>Large Break ROSA 983</th>
<th>Small Break ROSA 984</th>
<th>Steamline ROSA 952</th>
<th>FIST (1)</th>
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<tr>
<td>Water Level</td>
<td>5.06</td>
<td>5.21</td>
<td>5.14</td>
<td>14.15</td>
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<tr>
<td>Level 3</td>
<td>4.77</td>
<td>4.96</td>
<td>5.00</td>
<td>13.48</td>
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<tr>
<td>Level 2</td>
<td>4.60</td>
<td>4.46</td>
<td>4.76</td>
<td>12.19</td>
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<tr>
<td>Level 1</td>
<td>3.62</td>
<td>4.00</td>
<td>4.25</td>
<td>9.55</td>
</tr>
<tr>
<td>D.C. Bottom</td>
<td>0.49</td>
<td>0.49</td>
<td>0.49</td>
<td>3.35</td>
</tr>
</tbody>
</table>

#### B. Volumes from Downcomer Bottom (m³) (2)

<table>
<thead>
<tr>
<th>Level</th>
<th>Large Break ½ ROSA 983</th>
<th>Small Break ½ ROSA 984</th>
<th>Steamline ½ ROSA 952</th>
<th>FIST</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top of Vessel</td>
<td>0.3556</td>
<td>0.3556</td>
<td>0.3556</td>
<td>0.3925</td>
</tr>
<tr>
<td>Steamline</td>
<td>0.3556</td>
<td>0.3556</td>
<td>0.3556</td>
<td>0.2358</td>
</tr>
<tr>
<td>Top DP Tap</td>
<td>0.3556</td>
<td>0.2064</td>
<td>0.3556</td>
<td>0.1911</td>
</tr>
<tr>
<td>Water Level</td>
<td>0.1154</td>
<td>0.0973</td>
<td>0.1348</td>
<td>0.1072</td>
</tr>
<tr>
<td>Level 2</td>
<td>0.0460</td>
<td>0.0612</td>
<td>0.0712</td>
<td>0.0551</td>
</tr>
<tr>
<td>Level 1</td>
<td>0.0572</td>
<td>0.0572</td>
<td>0.0572</td>
<td>0.0504</td>
</tr>
<tr>
<td>Bottom DP Tap</td>
<td>0.0293</td>
<td>0.0293</td>
<td>0.0293</td>
<td>0.0369</td>
</tr>
<tr>
<td>Jet Pump Top</td>
<td>0.0050</td>
<td>0.0050</td>
<td>0.0050</td>
<td>0.0073</td>
</tr>
<tr>
<td>Recirc. Suction</td>
<td>0.0000</td>
<td>0.0000</td>
<td>0.0000</td>
<td>0.0000</td>
</tr>
</tbody>
</table>

(1) BWR elevations. Bottom of facility is 0.74 m BWR elevation.
(2) Includes volume between the separator cans and inner pipe.
### Table 18.4  Comparison of initial conditions and boundary conditions

<table>
<thead>
<tr>
<th></th>
<th>ROSA 983</th>
<th></th>
<th>FIST 6DBA1B</th>
<th></th>
<th>ROSA/FIST</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Total</td>
<td>Per Full Bundle</td>
<td>Total</td>
<td>Per Full Bundle</td>
<td>Total</td>
</tr>
<tr>
<td><strong>Initial Conditions</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure (MPa)</td>
<td>7.19</td>
<td></td>
<td>7.19</td>
<td></td>
<td>7.19</td>
</tr>
<tr>
<td>Power (MW)</td>
<td>3.62</td>
<td>1.81</td>
<td>5.05</td>
<td>0.36</td>
<td></td>
</tr>
<tr>
<td>Core Inlet Subcooling (K)</td>
<td>11.6</td>
<td></td>
<td>9.85</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core Inlet Flow (kg/s)</td>
<td>35.90</td>
<td>17.95</td>
<td>18.55</td>
<td>0.97</td>
<td></td>
</tr>
<tr>
<td>Feedwater Flow (kg/s)</td>
<td>1.26</td>
<td>0.63</td>
<td>2.45</td>
<td>0.26</td>
<td></td>
</tr>
<tr>
<td>Steam Flow (kg/s)</td>
<td>1.30</td>
<td>0.65</td>
<td>2.63</td>
<td>0.25</td>
<td></td>
</tr>
<tr>
<td>Water Level (m)</td>
<td>4.84(^{(1)})</td>
<td></td>
<td>14.15</td>
<td></td>
<td>1.08</td>
</tr>
<tr>
<td>Total Liquid Mass (kg)</td>
<td>600(^{(2)})</td>
<td>300</td>
<td>277(^{(3)})</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Boundary Conditions</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Power Trip (s)</td>
<td>9</td>
<td></td>
<td>0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Break Initiation (s)</td>
<td>0</td>
<td></td>
<td>0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Recirc. Suction Break (mm(^{2}))</td>
<td>539.</td>
<td>269.6</td>
<td>279.6</td>
<td>0.96</td>
<td></td>
</tr>
<tr>
<td>Drive Line Break (mm(^{2}))</td>
<td>80</td>
<td>40.1</td>
<td>51.8</td>
<td>0.78</td>
<td></td>
</tr>
<tr>
<td>Steamline Break (mm(^{2}))</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ADS Flow Area (mm(^{2}))</td>
<td>349.6</td>
<td>174.8</td>
<td>186.7</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ADS Trip (s)</td>
<td>115</td>
<td></td>
<td>None</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pump Trip (s)</td>
<td>0</td>
<td></td>
<td>0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feedwater Line Trip (s)</td>
<td>2-4</td>
<td></td>
<td>0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MSIV Trip (s)</td>
<td>13</td>
<td></td>
<td>L1+2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HPCS Trip (s)</td>
<td>27</td>
<td></td>
<td>27</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LPCS Trip (s)</td>
<td>50 &amp; 1.77 MPa</td>
<td></td>
<td>35 &amp; 1.97 MPa</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LPCi Trip (s)</td>
<td>50 &amp; 1.47 MPa</td>
<td></td>
<td>35 &amp; 1.68 MPa</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HPCS Temperature (K)</td>
<td>322</td>
<td></td>
<td>322</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LPCS Temperature (K)</td>
<td>322</td>
<td></td>
<td>322</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LPCi Temperature (K)</td>
<td>322</td>
<td></td>
<td>322</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HPCS Flow (kg/s)</td>
<td>0.80</td>
<td>0.40</td>
<td>0.59</td>
<td>0.68</td>
<td></td>
</tr>
<tr>
<td>LPCS Flow (kg/s)</td>
<td>1.38</td>
<td>0.69</td>
<td>0.72</td>
<td>0.95</td>
<td></td>
</tr>
<tr>
<td>LPCi Flow (kg/s)</td>
<td>1.38</td>
<td>0.69</td>
<td>0.52</td>
<td>1.32</td>
<td></td>
</tr>
<tr>
<td>Total ECCS Flow (kg/s)</td>
<td>3.54</td>
<td>1.77</td>
<td>1.83</td>
<td>0.97</td>
<td></td>
</tr>
<tr>
<td>Broken Loop Isolation (s)</td>
<td>N/A</td>
<td></td>
<td>0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Intact Loop Isolation (s)</td>
<td>N/A</td>
<td></td>
<td>13</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

\(^{(1)}\) Some uncertainty due to flow effects on the downcomer differential pressure measurement.

\(^{(2)}\) Total liquid mass estimated assuming downcomer liquid level at 5 m, mixture level inside the shroud to the top of the separator, and the slip ratio was 1.0.

\(^{(3)}\) Estimated mass.
<table>
<thead>
<tr>
<th>Key Events</th>
<th>Time (sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>ROSA 984</td>
</tr>
<tr>
<td>Break Valves Open</td>
<td>0.0</td>
</tr>
<tr>
<td>Pump Coast-Down Begins</td>
<td>0.0</td>
</tr>
<tr>
<td>Feedwater Line Closes</td>
<td>2.0 - 4.0</td>
</tr>
<tr>
<td>Jet Pump Suction Uncovers</td>
<td>12.0</td>
</tr>
<tr>
<td>Recirc. Suction Break Uncovers</td>
<td>17.0</td>
</tr>
<tr>
<td>Power Decay Initiated</td>
<td>8.0</td>
</tr>
<tr>
<td>MSIV Closes</td>
<td>13.0 - 15.0</td>
</tr>
<tr>
<td>Lower Plenum Flashing Begins</td>
<td>18.0</td>
</tr>
<tr>
<td>Guide Tube Flashing Begins</td>
<td>20.0</td>
</tr>
<tr>
<td>Lower Plenum Level Forms</td>
<td>27.1</td>
</tr>
<tr>
<td>Jet Pump Exit Uncovers</td>
<td>--</td>
</tr>
<tr>
<td>Jet Pump Loop Isolated</td>
<td>--</td>
</tr>
<tr>
<td>HPCS Flow Begins</td>
<td>27.0</td>
</tr>
<tr>
<td>Bundle Heat-Up Begins</td>
<td>39.6</td>
</tr>
<tr>
<td>PCT Occurs</td>
<td>81.2</td>
</tr>
<tr>
<td>PCT Temperature</td>
<td>575K</td>
</tr>
<tr>
<td>PCT Level Above BHL (Normalized by Heated Length)</td>
<td>0.81</td>
</tr>
<tr>
<td>LPCS Flow Begins</td>
<td>85.0</td>
</tr>
<tr>
<td>LPCI Flow Begins</td>
<td>95.0</td>
</tr>
<tr>
<td>ADS Actuation</td>
<td>115.0</td>
</tr>
<tr>
<td>Bypass/GT Refill Begins</td>
<td>--</td>
</tr>
<tr>
<td>Bypass CCFL Breakdown</td>
<td>--</td>
</tr>
<tr>
<td>Bypass Refilled</td>
<td>--</td>
</tr>
<tr>
<td>Bundle Reflooding Begins</td>
<td>92.3</td>
</tr>
<tr>
<td>Bundle Reflooded</td>
<td>114.6</td>
</tr>
<tr>
<td>UTP CCFL Breakdown</td>
<td>--</td>
</tr>
<tr>
<td>Bundle Quenched</td>
<td>120.8</td>
</tr>
<tr>
<td>Jet Pump Exit Recovers</td>
<td>--</td>
</tr>
</tbody>
</table>
Fig. 18.1 FIST flow diagram.
Fig. 18.2  Vessel regions.
Fig. 18.3 Axial power distribution.

Fig. 18.4 Facility elevations (m) and level instruments.
Fig. 18.5 Comparison of system pressures in ROSA-III and FIST tests.

Fig. 18.6 Comparison of mixture levels in downcomer in ROSA-III and FIST tests.
Fig. 18.7  Comparison of mixture levels inside core shroud in ROSA-III and FIST tests.
Fig. 18.8 Comparison of fuel rod surface temperatures in ROSA-III and FIST tests.

Fig. 18.9 Arrangement of heater rods and radial power distribution in FIST test facility.
Fig. 18.10 Nodalization diagram for ROSA-III analysis.

Fig. 18.11 Comparison of calculated and measured system pressures in ROSA-III.
Fig. 18.12 Comparison of calculated and measured fuel rod surface temperatures in ROSA-III.

Fig. 18.13 Comparison of calculated and measured system pressures in FIST.
Fig. 18.14 Comparison of calculated and measured fuel rod surface temperatures in FIST.

Fig. 18.15 Comparison of calculated system pressures in ROSA-III, FIST and BWR.
**Fig. 18.16** Comparison of calculated fuel rod surface temperatures in ROSA-III, FIST and BWR.
19. BWR Small Break LOCA Counterpart Tests at ROSA-III and FIST

19.1 Introduction

The Japan Atomic Energy Research Institute (JAERI) and the General Electric Company performed 2.8% recirculation pump suction line break tests at the ROSA-III and FIST facilities as counterpart tests, respectively\(^1\)\(^{-6}\). The objective of the tests was to develop a common understanding and interpretation of the controlling phenomena during BWR small break LOCAs through comparisons of the test results from different test facilities with different scaling concepts. The FIST facility is a 1/624 volumetrically scaled model of a BWR/6-218 with 624 fuel bundles. The facility is characterized by the use of a full-height vessel and a single full-length bundle. Detailed information about the FIST facility is included in Chapter 18.

Using data from these tests, the THYDE-BI\(^7\) code, a fast-running code developed by JAERI for best-estimate calculation of BWR small- and intermediate-break LOCAs, was assessed. Then by using this code, the results obtained in both tests were extrapolated to a BWR, and the BWR small break LOCA was investigated.

19.2 Test Conditions

After the initial test conditions were established, a LOCA test was initiated. The test conditions of both tests are summarized in Table 19.1. The HPCS was not used in either test. This represented a failure of the diesel generator powering the HPCS.

19.3 Comparison of Test Results

The sequence of events in both tests is similar, although minor differences exist.

The vessel pressure transients are shown in Fig. 19.1. After the break occurs, the pressure control system is activated and the vessel pressure is kept constant. After the MSIV is closed, the vessel pressure increases. The downcomer liquid level decreases because of fluid effluent from the break. This results in a signal activation which trips on the ADS after a time delay of 120 s. After the ADS actuation the vessel pressure decreases rapidly. Flashing occurs in the lower plenum and downcomer at 6.4 MPa. As the flashing becomes less effective and the core fluid boils off, the core mixture level begins to fall. The entire core finally uncovers, as shown in Fig. 19.2. The cladding surface temperature excursion propagates downward from the top to the bottom of the core as the core mixture level decreases, as shown in Figs. 19.3 and 19.4. The rapid depressurization leads to the actuation of the LPCS and the LPCI system. Quenching propagates downward from the core top after the LPCS initiation, and also upward from the bottom part of the core as the core begins to be refilled. The core is reflooded after the LPCI initiation.

Some differences between the ROSA-III and FIST tests are observed in the pressure transient from the break initiation through the MSIV closure in Fig. 19.1. In the ROSA-III test, the pressure control system was set to prevent the vessel pressure from decreasing below 6.7 MPa. However, in the FIST test, the pressure control system was set to maintain the vessel pressure at the initial pressure of 7.23 MPa even if a small break occurred. Thus, in
the ROSA-III test, the pressure decreases to 6.7 MPa after break and is maintained at 6.7 MPa until the MSIV closure. In the FIST test, the pressure is maintained at the initial value. Therefore, this difference stemmed from the difference in experiment specifications between the two tests and not from any scaling difference. It is concluded that the difference in pressure transients early in the tests is insignificant in the scope of the counterpart experiment objectives. However, the effect on the transients later in the tests is large, as explained below.

The ADS actuation in the ROSA-III test is approximately 50 s later than in the FIST test, because the downcomer level decrease in the ROSA-III test is slower than in the FIST test. This is due to ROSA-III's larger initial downcomer water inventory as mentioned in Chapter 18 and lower setpoint pressure for the pressure control system which results in a smaller mass loss from the break during the pressure control system operation.

The depressurization after the ADS actuation is slower in the ROSA-III test than in the FIST test, because an oversized ADS flow area was used in the FIST test to compensate for the stored energy release from the structure. No concern was paid to the stored energy effect in the ROSA-III test. The larger total initial water mass in the ROSA-III test is another probable reason for the slower depressurization after the ADS actuation.

The inside-shroud mixture level swell due to flashing initiation after the ADS actuation in the ROSA-III test is much smaller than that in the FIST test as shown in Fig. 19.2. Two full-length bundles are simulated by four half-length bundles in the ROSA-III core, which results in a larger core flow area in the ROSA-III test facility than that in the FIST facility. The upper plenum and steam separator in the ROSA-III test facility are shortened and widened compared with those in the FIST facility, which results in larger flow areas in those regions in the ROSA-III facility. The larger flow area in the ROSA-III facility lets vapor generated in the mixture volume leave from the mixture level more rapidly than in the FIST facility. Thus, the inside-shroud mixture level swell after the ADS actuation is smaller in the ROSA-III test than that in the FIST test. The slower depressurization after the ADS actuation in the ROSA-III test than that in the FIST test also has an impact on the difference in the inside-shroud mixture level swell between the two tests.

No mixture level is formed in the lower plenum in the ROSA-III test, whereas a mixture level is formed in the FIST test after the lower plenum flashing (LPF) initiation, as shown in Fig. 19.2. In the ROSA-III test, an increase in downcomer water inventory after the LPF initiation is not clearly observed as shown in Fig. 19.5. In the FIST test, an increase in downcomer water inventory and decrease in inside-shroud water inventory after the LPF initiation are clearly observed as shown in Fig. 19.6. This implies that the effluent of water from the lower plenum to the downcomer after the LPF initiation is not prominent in the ROSA-III test and it is quite prominent in the FIST test. The conduction probes in the upper part of the downcomer in the FIST test facility did not show the existence of water there in this time period. Therefore, it is believed that water in the lower plenum is pushed into the downcomer through the jet pumps by vapor generated by the LPF, and falling water from the separator to the downcomer is a small amount. As a result, a mixture level is formed in the lower plenum in the FIST test because of counter current flow limiting (CCFL) holdup of the core coolant mixture.

One probable reason for the above difference is the difference of side entry orifice scaling between the two facilities. In the ROSA-III facility, the differential pressure across the side entry orifice at steady state conditions is preserved. Thus, the diameter of the ROSA-III side entry orifice is $1/\sqrt{2}$ times as large as that of the BWR since the flow rate per half-length bundle in the ROSA-III is half of the flow rate per full-length bundle in the BWR and FIST. There is also another flow path between the lower plenum and the core in the
ROSA-III facility. There is a clearance between the electrode of a simulated fuel rod and the core bottom plate. The clearance width is 0.42 mm and the total area is 1060 mm². Therefore, the flow velocity at the side entry orifice may be smaller than expected, approximately 60% of the planned value. It is suspected that this distorted the CCFL phenomena at the side entry orifice in the ROSA-III test and water easily falls from the core to the lower plenum. In the FIST test, the CCFL phenomena is scaled at the side entry orifice and a mixture level is formed in the lower plenum. The occurrence of this CCFL phenomena increases the flow resistance from the lower plenum to the steam dome through the core, and vapor continuously generated in the lower plenum pushes water into the downcomer through the jet pumps.

Another probable reason is the difference in the depressurization rate between the two tests. The depressurization after the ADS actuation in the FIST test is faster than that in the ROSA-III test. Thus, the vapor velocity at the side entry orifice in the FIST test is higher than that in the ROSA-III test. This creates a difference in the CCFL liquid holdup at the side entry orifice observed between the two tests.

It is difficult to compare the mixture level swell in the downcomer between the two tests, shown in Figs. 19.7 and 19.8, since there are no data from 300 to 400 s in the ROSA-III test. However, it can be observed that the mixture level swell after the flashing initiation in the FIST test is larger than that in the ROSA-III test. One probable reason is that the downcomer of the ROSA-III facility is shorter and wider than that of FIST and the mixture level swell is smaller. Another reason is that the downcomer of the ROSA-III facility is an annulus like a BWR’s downcomer, whereas the downcomer of the FIST is a pipe. This effect will be discussed more in Section 19.4.2(2).

The mixture level recovery inside the shroud and in the downcomer occurs earlier in the FIST test than in the ROSA-III test, because the depressurization occurs sooner and the LPCS and the LPCI begin to inject water earlier in the FIST test than in the ROSA-III test, respectively.

The peak cladding temperature (PCT) occurred near the core midplane in both tests. The PCTs were 710 K in the ROSA-III test and 769 K in the FIST test, respectively.

Multichannel phenomena, such as the coexistence of upward and downward flows in the core, was observed for a brief time after the lower plenum flashing initiation. However, all four ROSA-III bundles responded similarly in the dryout propagation period.

It is confirmed from the above observations that all the phenomena observed in both experiments are mutually common and there are no conflicts. Although some differences are observed in the extent or timing of the phenomena, these are attributable to unintentional differences in the experiment specifications or to facility scaling differences. The facility scaling difference which shows the most significant effect is the flow area. The flow area scaling has a significant effect on the two-phase mixture level swell due to flashing and the CCFL phenomena. In the ROSA-III facility, each region is shortened and widened, and volumetrically scaled. In the FIST facility, the height of each region is preserved and the facility consists of narrow flow paths. In a BWR, the volume and flow area of each region are much larger than that of the ROSA-III or FIST facility. Two-phase flow characteristics are very sensitive to these factors and it is supposed that even if steam and vapor mass fluxes are preserved, two-phase flow phenomena in a large flow path and a small flow path would be different. It is concluded that the important results obtained from the present counterpart experiments are not which test results are closer to the BWR LOCA situation, but that no conflicting phenomena are observed in either experiment. Therefore, if both test results are predicted reasonably by a LOCA analysis computer code, the code should be applicable.
to a facility with a different scaling concept or to a BWR.

19.4 Analysis

Test results of ROSA-III and FIST were analyzed by the THYDE-B1 code\textsuperscript{79}. A 2.8\% BWR LOCA was also analyzed by the same code and the BWR small break LOCA phenomena was investigated.

19.4.1 Code Description and Modification

The THYDE-B1 code is a fast-running computer code developed for analyzing the thermal-hydraulic response of a BWR during a LOCA. The code is a one-dimensional lumped-parameter code with use of a coarse nodalization.

The three-region node model, unique to the code, can consider up to three vertically-stacked subnodes which represent a saturated vapor region, a saturated two-phase mixture region and a subcooled liquid region, respectively.

The vapor separation rate from the mixture which is important for calculation of the mixture level within a three-region node is calculated from the void fraction and the bubble rise velocity. In calculating the void fraction, bubble sweep-out lengths for flashing-generated vapor and heat-transfer-generated vapor are used. The depth of the mixture region is used for the bubble sweep-out length for the flashing-generated vapor in the present analyses, because the flashing due to rapid depressurization inherently occurs in the entire region in the vessel. The half length of the heat slab facing the mixture region is used for the bubble sweep-out length for the heat-transfer-generated vapor. The bubble rise velocity is calculated from the Wilson correlation\textsuperscript{80} in the original code. The Wilson correlation was derived from experiments in circular tubes. It usually gives a lower bubble rise velocity in rod bundles. Thus, the Cunningham-Yeh correlation\textsuperscript{81} which was derived from experiments in a rod bundle is used for the core bubble rise velocity in the present analyses to get better predictions. The Wilson correlation is still used in other regions.

A post-dryout heat transfer coefficient of 40 W/m\(^2\)K for the period before the initiation of the LPCS and 70 W/m\(^2\)K for the period thereafter until reflooding were used based on the experiences of ROSA-III experiment analyses\textsuperscript{10}.

Figure 19.9 shows the nodalization diagram for the ROSA-III test analysis. The nodalizations for the FIST test and BWR small break LOCA analyses are very similar to it.

The discharge flows through the break, the main steam line, the safety relief valve (SRV) and the ADS were calculated using the Bernoulli equation with a discharge coefficient of 0.60 for the subcooled upstream condition, and with the Moody-slip critical flow model\textsuperscript{11} with a discharge coefficient of 0.61 for the saturated upstream condition, respectively.

19.4.2 Analytical Results for ROSA-III and FIST Tests

The overall behavior of both tests is successfully reproduced by the THYDE-B1 code.

(1) Vessel Pressure

The THYDE-B1 code calculates the vessel pressure transients in the ROSA-III and FIST tests well, as shown in Fig. 19.1. The predicted timings of the MSIV closure and the ADS actuation for both tests are very close to that measured. The events depend on the downcomer collapsed level transient. Thus, it means that the code could calculate the downcomer collapsed level transient properly as will be discussed later.

The calculated pressurization after the MSIV closure is a little faster than that measured in both tests. It is attributed to the lack of condensation heat transfer model in the code.
Consensation of vapor on the vessel structure should have occurred when the vessel was pressurized, and must have influenced the pressurization rate.

The depressurization after the ADS actuation primarily depends on the balance between the heat generation in the core and the energy and mass release from the vessel. The agreement between the ROSA-III data and the predicted depressurization is quite good. It suggests that the Moody critical flow model with a discharge coefficient of 0.60 for the calculation of the break and ADS flow rates is adequate. The predicted FIST depressurization is slightly faster than in the test, which results in a slightly earlier actuation of the LPCS and the LPCI.

(2) Downcomer Level

The downcomer level transient is important because trip signals such as the MSIV closure, the ADS opening, and the LPCS and LPCI actuation are generated by the downcomer level.

The predicted and measured downcomer levels are compared in Fig. 19.7 for the ROSA-III test and in Fig. 19.8 for the FIST test. The code predicts the downcomer level decrease after the break initiation quite well. The downcomer fluid is single-phase liquid until the ADS actuation and the downcomer level is primarily dependent on the break flow. It is implied again that the THYDE-B1 discharge flow calculation model used in the present analysis is adequate. It is also implied that the THYDE-B1 code can calculate the trip signal activation properly.

The agreement of the predicted and measured mixture levels after the flashing initiation due to the ADS actuation is poor in the ROSA-III case, whereas it is good in the FIST case. The reason is attributed to the Wilson bubble rise velocity correlation. The FIST downcomer is simulated by a circular tube located outside the vessel. The hydraulic diameter is approximately 10 cm. Circular tubes with hydraulic diameters from 10 cm through 48 cm were used in the Wilson experiments to derive the correlation. Thus, the prediction of the FIST downcomer mixture level is satisfactory. The ROSA-III downcomer is annular like the BWR downcomer. Bubbles produced by flashing coalesce into large bubbles in the annular flow channel. The equivalent diameter of the coalesced bubble sometimes becomes larger than the hydraulic diameter of the annular flow channel. Thus, Wilson correlation based on the circular tube data cannot be applied to the calculation of the bubble rise velocity of large bubbles in the annular flow channel. Therefore, the agreement of the measured and calculated downcomer mixture levels after the flashing initiation is poor in the ROSA-III case. This explanation will be confirmed later with the region mass calculation results. It is not sure that the THYDE-B1 code can accurately predict the mixture level transient in the large annular downcomer of a BWR. However, the downcomer mixture level transient after the flashing initiation is not so important as before the flashing initiation. Thus, the poor mixture level calculation in the downcomer has minor effect on the system response.

(3) Inside-Shroud Mixture Level

Prediction of the core mixture level is important because the rod temperature closely responds to the mixture level behavior, particularly during the core uncover process.

The predicted core mixture levels for ROSA-III and FIST cases and the test data are compared in Fig. 19.2.

In the ROSA-III case, the THYDE-B1 code can accurately reproduce the core mixture level decrease. However, the entire core dry-out is not calculated and the mixture level recovery after the LPCS and LPCI actuation is a little slower in the calculation than in the test.

In the FIST case, the agreement of the predicted and measured inside-shroud mixture levels is good except in the lower plenum. The mixture levels are formed at the same time period in the core and the lower plenum in the test because of the CCFL occurrence at the
side entry orifice of the core. The THYDE-B1 code cannot handle this phenomena because it is not modeled. Even so, the code can properly calculate the core mixture level fall before the lower plenum flashing, the mixture level swell due to the lower plenum flashing, the core mixture level boil-off and the core mixture level recovery after the LPCS and LPCI initiation. The spike in the calculated core mixture level after the LPCS actuation is due to a numerical problem caused by abrupt flashing when water comes into contact with dry heat slabs.

It is concluded that the THYDE-B1 code can accurately predict the core mixture level transient for a small break LOCA by using the Cunningham-Yeh’s bubble rise velocity.

(4) Residual Mass Inventory Distribution

Figures 19.5 and 19.6 provide comparisons of predicted and measured residual downcomer and inside-shroud mass inventory transients for the ROSA-III and FIST tests, respectively.

The agreement between the predicted and measured ones is excellent in both cases. In contrast, as shown in Fig. 19.7, the downcomer mixture level transient after the flashing initiation was not well predicted in the ROSA-III case. However, in terms of the residual downcomer mass inventory transient, the THYDE-B1 code provides a good prediction for the ROSA-III case. This confirms the need for an adequate mixture level calculation model for the annular downcomer.

Large mass inflow from the inside-shroud into the downcomer after the LPF initiation is observed in the FIST test. It causes a mixture level formation in the lower plenum as shown in Fig. 19.2. The THYDE-B1 code can predict the lower plenum-to-downcomer mass transfer adequately. The THYDE-B1 code cannot predict the CCFL phenomenon across the side entry orifice at the core inlet. Thus, it cannot predict the level formation in the lower plenum as presented in Fig. 19.2. However, the THYDE-B1 code can calculate the residual mass transients in the downcomer and inside the shroud surprisingly well.

The THYDE-B1 code primarily considers only the pressure balance to calculate the mass distribution in the system. It is confirmed from Figs. 19.5 and 19.6 that if the mass leakage from the system is calculated properly the THYDE-B1 hydraulic model is sufficient to provide a good prediction for the mass distribution in the system.

(5) Cladding Temperature

The predicted cladding temperatures are compared with measured ones in Figs. 19.3 and 19.4 for the ROSA-III and FIST cases, respectively. It should be noted that the thermocouples which measure the rod temperatures in the FIST test are embedded in the ceramic cement which fills the interior of the heater rod while the predicted cladding temperature is the outer surface temperature of the heater rod. Thus, there are differences when directly comparing the predicted cladding temperature with the measured.

The timing of the cladding temperature excursion start and the quenching, and the heat transfer coefficient during the uncovered period are important for cladding temperature prediction. The occurrence of the dryout is determined by the core mixture level fall and the timing of the quenching is determined by the core mixture level recovery caused by the injection from the LPCS and the LPCI. The injection times are primarily determined by the vessel pressure transient.

Except for the lower part of the core, the timing of the dryout is predicted well by the THYDE-B1 code in the ROSA-III case. As previously mentioned in Section 19.4.2(4), the THYDE-B1 code provided a good prediction for the ROSA-III core mixture level fall except for the lower part of the core. The quenching time is well predicted except for the lower part of the core, because the pressure transient is well predicted. The predicted cladding
heatup rates are very close to the measured ones until the LPCS initiation. However, the THYDE-B1 code gives a little lower cladding temperature after the LPCS initiation than measured in the experiment. It appears that the model heat transfer coefficient of 40 W/m²K for the dryout period before the LPCS initiation is adequate. However, the coefficient of 70 W/m²K used for the period thereafter until reflooding seems a little higher than that in the present experiment. Even so, the predicted cladding temperatures are very close to the measured ones as a whole.

The THYDE-B1 code predicts the timing of the cladding temperature excursion start very well for the FIST case because the core mixture level fall is well predicted, shown in Fig. 19.2. After dryout occurs, the difference between the temperature measurement locations and the predicted locations may be reflected in the results making a direct comparison difficult.

The predicted peak cladding temperatures for the ROSA-III and FIST case are 677 K and 633 K, respectively.

It is concluded that the THYDE-B1 code can predict the timing of cladding temperature excursion start and quenching satisfactorily. The model heat transfer coefficients are adequate for the dryout period and a little large for the LPCS actuation period. It is confirmed that the THYDE-B1 code is capable of providing an adequate prediction for the cladding temperature transient during a BWR small break LOCA.

19.4.3 BWR Analysis Results

(1) Vessel Pressure

The predicted BWR, ROSA-III and FIST pressure transients are compared in Fig. 19.10. In the BWR analysis, the pressure control system was set to regulate the vessel pressure at 6.7 MPa which was the same as the value used in the ROSA-III test. In the FIST test, the pressure setting of the pressure control system was 7.23 MPa, the pressure at steady state. Thus, the BWR pressure transient is close to the ROSA-III results in the early portion of the transient. The initial downcomer liquid mass in the ROSA-III test was larger than in the BWR case. Thus, the L1 downcomer liquid signal was activated a little earlier in the BWR case than in the ROSA-III case. This cause a slightly earlier MSIV closure and ADS actuation in the BWR case than in the ROSA-III case.

The LPCS and LPCI initiation times in the BWR case are also a little earlier than in the ROSA-III case. The initiation signals are activated by the L1 signal. However, the actual injections are initiated when the vessel pressure falls below the shut-off head of each pump. The BWR vessel pressure after the ADS actuation is a little lower than the ROSA-III results because the ADS actuation was slightly earlier.

(2) Downcomer Level

The predicted downcomer levels for the BWR, ROSA-III and FIST are presented in Fig. 19.11. The downcomer heights of the three are different, thus the downcomer height is normalized by the initial liquid level L3 and L1 level.

The BWR downcomer liquid level transient is close to the ROSA-III result except for the early portion of the transient. The level reaches the L1 level a little earlier than in the ROSA-III case. After the flashing initiation, caused by rapid depressurization due to the ADS actuation, the BWR downcomer mixture level swells. However, the ROSA-III level swell predicted is considered to be incorrect because the THYDE-B1 mixture level calculation model cannot adequately predict the mixture level in the narrow annular downcomer, as discussed in Section 19.4.2(2).
(3) Inside-Shroud Mixture Level

The BWR inside-shroud mixture level is shown in Fig. 19.12. The predicted ROSA-III and FIST inside-shroud mixture levels are also shown for comparison. The levels are normalized by the height of the upper plenum, core and lower plenum, respectively.

The uncovering of the upper part of the core before the LPF initiation is predicted in the BWR case. It is also observed and predicted in the FIST test. The BWR core is filled again with a two-phase mixture after the LPF. The BWR core uncovering occurs again as flashing surge decreases and the core fluid boils off. The second core uncovering is very similar to the ROSA-III case. Entire core uncovering occurs neither in the BWR case nor in the ROSA-III case. The core is reflooded by the LPCS and LPCI injections as also occurred in the ROSA-III and FIST calculations.

(4) Residual Mass Inventory Distribution

The BWR downcomer and inside-shroud mass inventory transients are presented in Fig. 19.13. The predicted ROSA-III and FIST mass inventory transients are also shown for comparison. The mass inventories are standardized by converting to per-one-bundle values in order to make the comparison easy.

The BWR downcomer mass inventory transient is very close to that of the FIST case until the LPF initiation. Thereafter, it is close to that of the ROSA-III case, because the lower plenum-to-downcomer liquid mass transfer is not predicted prominently in the BWR case. The mass transfer was observed in the FIST test, whereas it was not clear in the ROSA-III test. The depressurization after the ADS actuation in the BWR case is very similar to that of the ROSA-III case and slower than that in the FIST case. This may be the reason for the difference in the lower plenum-to-downcomer liquid mass transfer between the BWR case and the FIST case. The FIST downcomer mass recovery is earlier than in the BWR and the ROSA-III cases, because the LPCS and the LPCI begin injection earlier.

The inside-shroud mass inventory transients of the three cases are very similar except that the lower plenum-to-downcomer liquid mass transfer is clearly calculated only in the FIST case. This was observed in the FIST test also. The FIST inside-shroud mass inventory recovery is earlier than the BWR and the ROSA-III, because the LPCS and the LPCI actuations are earlier.

(5) Cladding Temperature

The predicted cladding temperatures for the BWR, ROSA-III and FIST cases are compared in Fig. 19.14.

The temporary dryout at the upper part of the core is predicted before the LPF in the BWR case, and was observed and predicted in the FIST test. The rods are rewetted by the mixture level swell caused by the LPF.

The timing of the cladding temperature excursion start after the abatement of the LPF in the BWR case is very close to those in the ROSA-III case, as expected from the core mixture level transients shown in Fig. 19.12. The cladding temperature excursion occurs only above the core mid-plane. The cladding temperature heat-up rates in the BWR case are a little higher than in the ROSA-III case. The quenching occurs a little earlier in the BWR case than in the ROSA-III case because the LPCS and the LPCI actuations are a little earlier as shown in Fig. 19.10. The predicted BWR peak cladding temperature is 697 K which is much lower than the licensing criterion of 1473 K. It was pointed out in Section 19.4.2(5) that the THYDE-B1 code with the post-dryout heat transfer coefficient of 70 W/m²K for the LPCS spray period tends to underpredict the cladding temperature. However, the underprediction is within 33 K in the ROSA-III prediction as shown in Fig. 19.3, and negligible compared with the large safety margin in the PCT.
19.5 Conclusions

The following conclusions were obtained from the ROSA-III and FIST test results and the calculational results for the ROSA-III, the FIST and the reference BWR using the THYDE-B1 code.

1) The basic sequence of events and the key phenomena in the two small break LOCA tests at the ROSA-III and the FIST facilities were similar. The PCTs in both tests occurred at the midplane of the core and were nearly the same in magnitude.

2) Although some differences between the ROSA-III and FIST test results were observed in the extent or timing of the phenomena, they were attributable to the scaling difference except for the experiment specification difference. The scaling difference which showed the most significant effect is the flow area. The effect of this scaling difference appeared in the difference of the two-phase mixture level swell caused by flashing due to rapid depressurization.

3) The THYDE-B1 code reproduced both test results quite well. The analyses showed that the code has the capability to accurately predict thermal-hydraulic responses during a small break LOCA of a BWR. It is recommended to improve the calculation models for the annulus downcomer mixture level and the post-dryout heat transfer coefficient during the LPCS spray actuation for better predictability.

4) It was confirmed through the analyses that the fundamental phenomena observed in both tests are similar to those during a BWR small break LOCA.

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<table>
<thead>
<tr>
<th>Item</th>
<th>ROSA-III</th>
<th>FIST</th>
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<tr>
<td>Initial Conditions</td>
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<tr>
<td>Pressure (MPa)</td>
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<td>Power (MW)</td>
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<td>Core inlet flow (kg/s)</td>
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<td>Downcomer water level (m)</td>
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<td>Total liquid volume (m³)</td>
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<tr>
<td>Break area (mm²)</td>
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<td>7.44 (2.8%)&lt;sup&gt;(2)&lt;/sup&gt;</td>
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<td>ADS flow area (mm²)</td>
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<td>Power trip (s)</td>
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<td>Feedwater trip (s)</td>
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<tr>
<td>MSIV trip (s)</td>
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<td>L1&lt;sup&gt;(3)&lt;/sup&gt;</td>
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<td>LPCS trip (s)</td>
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<tr>
<td>LPCI trip (s)</td>
<td>L1 + 40</td>
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</tr>
<tr>
<td>ADS trip (s)</td>
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<td>L1 + 120</td>
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<td>Pump trip (s)</td>
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<tr>
<td>Loop isolation (s)</td>
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<tr>
<td>LPCS pump shut-off pressure (MPa)</td>
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<tr>
<td>LPCI pump shut-off pressure (MPa)</td>
<td>1.57</td>
<td>1.68</td>
</tr>
</tbody>
</table>

Note: (1) Reference systems are a little different.
(2) L1 is 4.00 m in ROSA-III and 8.82 m in FIST, respectively.

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20. BWR Large Steam Line Break LOCA Counterpart Tests at ROSA-III and FIST

20.1 Introduction

The JAERI performed a 100% main steam line break test (Run 952)\textsuperscript{1} at the ROSA-III facility, and the General Electric Company also performed a similar steam line break test (6MSB1) at the FIST facility\textsuperscript{2}.

The major test conditions are similar between the two tests, except for some test conditions, such as the core power curves and ECCS actuation logics which differ because these two tests were not planned originally as counterpart tests.

To perform the similarity evaluation of the 100% steam line break LOCA between the ROSA-III, FIST and BWR/6 (848), the following studies were performed\textsuperscript{3}: (1) a study of the controlling phenomena in the large steam line break LOCAs in the ROSA-III and FIST tests\textsuperscript{4} and (2) an analytical study of the similarity of the major events with the same scaled analytical conditions in the ROSA-III, FIST and BWR/6 (848) systems. The RELAP5/MOD1 (Cy 18) code\textsuperscript{5} with a jet pump model\textsuperscript{6} developed in JAERI is used for (a) the post-test analyses of the two tests for code assessment and for establishing analytical conditions and (b) the similarity analyses of the large steam line break LOCAs at the three different systems.

20.2 Test Description

20.2.1 Test Facilities

The scaling objectives of ROSA-III and FIST facilities (cf. Fig. 20.1) are to simulate, on a real time basis, BWR system thermal-hydraulic response following a postulated LOCA. The important BWR system features that govern the mass and energy transfer rates are also preserved (see Table 18.1). These include vessel internals, channeled fuel bundles, bypass region, pump loops, feedwater and steam systems, water level trip logics and emergency core cooling systems (ECCs). The ROSA-III facility for the main steam line break tests is described in Chapter 15.

The full-height vessel of the FIST facility, with a single bundle core and volumetrically scaled flow areas, provides full scale values for the static heads. The two main recirculation loops are used to establish the initial test conditions and are isolated after coastdown of the recirculation pump so that the excess mass in the loops does not affect the transient phenomena. The feedwater lines are also isolated immediately after the break. The steam dryer and steam dome of the FIST facility simulate those of the reference BWR. The downcomer water level is measured outside the dryer skirt by a differential pressure transducer and is used for the trip signals.

Table 20.1 compares the metal volume including fuel rods and metal surface areas contacting the primary fluid in the ROSA-III and FIST systems. It shows that the metal structures of ROSA-III and FIST have the same metal surface area per full-size bundle and that the former has thicker walls.

20.2.2 Test Conditions

The initial and transient test conditions (Table 20.2) show the similar steam dome pressure and enthalpy distribution across the core in the ROSA-III and FIST tests, which are
close to the BWR rated conditions. However, the initial mass inventories in the two tests calculated by the RELAP5/Mod1 code deviate from the volumetrically scaled fluid mass.

The main steam isolation valve (MSIV) in FIST was tripped to close at 5.5 s after the break and thereafter simulated only the steam discharge flow from the broken steam line. On the other hand, the trip of MSIV closure in the ROSA-III test was assumed to occur at the same time as the break.

The transient core power curves, per full-size bundle, used in the ROSA-III and FIST tests are shown in Fig. 20.2. The transient heat flux in ROSA-III becomes higher than that of FIST at 5.5 s after the break and agrees well with that of FIST at 300 s after the break.

The important difference between the ROSA-III and FIST test conditions is in the ECCS failure mode. In the ROSA-III test, the high containment pressure signal was assumed to fail and all the ECCSs were expected to be tripped by the downcomer water level signals as shown in Table 20.2. On the other hand, a failure of two LPCIs was assumed in the FIST test and a high containment pressure signal was assumed to be sent at the time of break. Through the comparison of both test results, the effects of each ECCS on the steam line break LOCA were investigated.

20.3 Comparison of Test Results

20.3.1 Pressure Response and Major Events

Major events in both tests are compared in Table 20.3. After break initiation in the main steam line, the steam dome pressures in the two tests began to decrease rapidly as shown in Fig. 20.3. The lower plenum flashing began at 5 s and 4.2 s after the break in FIST and ROSA-III tests, respectively. The later lower plenum flashing initiation in FIST is due to lower core inlet fluid temperature in the initial condition. After lower plenum flashing, the depressurization rates of both tests slowed and thereafter both pressures decreased smoothly. The large liquid mass in the vessel contributed to decrease the depressurization rate later in the blowdown phase. This is typical of the large steam line break LOCA or of a top break LOCA and is distinctly different from the 200% main recirculation line break (MRLB) LOCA7 where a large amount of fluid mass is lost during the early blowdown phase as shown in Fig. 20.3.

The reason for the slight pressure difference between the two tests can be ascribed to (1) the faster depressurization in FIST due to the larger steam discharge flow during the first 5.5 s and actuation of three ECCSs, and (2) the slower depressurization in ROSA-III due to the larger initial mass inventory (see Table 20.2) and initiation of feedwater flashing at 2.2 MPa.

The HPCS in ROSA-III test was actuated at 94 s after the break (at 2.3 MPa) by the L2 trip signal with a time delay of 23 s. That in FIST was actuated at 27 s after the break. The LPCS and one LPCI in the FIST test were actuated at 88 and 95 s, respectively at their designed pressures.

20.3.2 Downcomer Water Level Transient during Depressurization

Figure 20.4 shows the relative mixture level transients. The levels were normalized by the distance between the initial water level and loop nozzle in the downcomer. In FIST, the mixture level showed that the main steam line was covered by the swelled mixture at 6 s to 80 s after the break. Another mixture level was observed below the orifice at the top of dryer skirt at 46 s due to the counter current flow limiting (CCFL) at the orifice. A mixture level was also detected in the downcomer bend pipe at 8 s after the break, indicating the
occurrence of CCFL at the top of the pipe bend, and diminished at 80 s after the LPCI actuation. The CCFL observed in the FIST test may not be observed in the BWR system in the same manner because the BWR downcomer is not one-dimensional as in the FIST downcomer.

On the other hand, no mixture level was detected by the conduction probes in the ROSA-III lower downcomer. The data of differential pressure in the lower downcomer region (elevation 0.94 m to 3.9 m above the PV bottom) was converted to void fraction by neglecting the frictional pressure loss in the measuring region. However, there was no data to estimate the void fraction in the upper downcomer. The upper downcomer mixture level in Fig. 20.4 was estimated from the data of collapsed water level above the 3.9 m elevation by assuming the same void fraction as in the lower downcomer. The estimated mixture level covered the main steam line (top of steam dome) at 15 s to 31 s after the break.

Thus, the rapid swelling of the mixture level in the downcomer and steam dome are commonly observed in both tests. However, the mixture level height and time period of steam line coverage by swelled mixture, are strongly influenced by the height and geometry of the downcomer and steam dome in each test facility.

20.3.3 Core Cooling Phenomena

Figure 20.5 shows representative heater rod surface temperatures at the seven axial elevations in the ROSA-III and FIST cores. ROSA-III fuel rods A-22 in the high power bundle and C-22 in the average power bundle have the same local peaking factor of 1.0, and radial peaking factors of 1.4 and 1.0, respectively. All the heater rods in the ROSA-III test showed a temperature increase due to the mixture level fall and subsequent quenching from the bottom by the HPCS actuation at 94 s after the break. The mixture level detected by the conduction probes in the ROSA-III test agreed well with the movement of dryout and quench fronts of heater rods. Slightly different temperature responses were observed between the high and average power bundles of ROSA-III, i.e., earlier dryout, lower rate of temperature increase after dryout and later quench in the average power bundles. On the other hand, none of the heater rods in FIST test showed any temperature increase, indicating sufficient mass inventory to cool the bundle. The difference in surface temperature responses between the two tests can be ascribed to the higher power generation rate at 5.5 s after the break and later, later actuation of HPCS, and no LPCS and LPCI actuations in the ROSA-III test.

20.4 Similarity Analyses of ROSA-III, FIST and BWR/6

20.4.1 Analytical Models

The RELAP5/MOD1 (CY18) code with the JAERI Jet Pump Model9) was used to perform the post test analyses for the ROSA-III (Run 952) and FIST (6MSB1) tests, and the similarity analyses for the same 100% steam line break LOCA with a 2-LPCI-failure assumption in ROSA-III, FIST and BWR/6 (848) systems.

In the post test analyses, the ROSA-III and FIST systems were represented by the nodings shown in Figs. 20.6 and 20.7, respectively. The initial and transient test conditions were correctly modeled in each analysis as shown in Table 20.4. In these analyses, non-homogeneous models were commonly applied to each component of the ROSA-III and FIST systems. A discharge coefficient of 0.5 for the steam line break flow rate was obtained from the results of post-test analysis8) for Run 952 conducted previously using the RELAP5/MOD1 (CY1) code. It was concluded through the post test analyses for ROSA-III and FIST that the RELAP5 code was capable of reproducing the major thermal-hydraulic phenomena for both steam line break tests.
By using the results of post test analyses, the similarity analyses were performed for the 100% steam line break LOCA in the ROSA-III, FIST and BWR/6 systems. To investigate the similarity of the major thermal-hydraulic phenomena, the same trip logic was used for the MSIV, feedwater system and ECCS, and volumetrically scaled injection flow rates and initial mass inventories were assumed. A 2-LPCI-failure assumption was used in those analyses. Major analysis conditions were established as shown in Table 20.4. The total fluid volume and initial mass inventory in Table 20.4 do not include the hot fluid remaining in the feedwater line of the ROSA-III and BWR/6 systems.

Based on these assumptions, the LPCS and LPCI were added to the similarity analysis for ROSA-III (case ROSA-S) and they were actuated at the same pressures as those of BWR. The excess initial water mass in the downcomer in the post-test analysis was removed in this calculation accompanying a volume remodeling for the component 111 as shown in Fig. 20.6. In addition to this calculation for ROSA-III, another case (case ROSA-M) was calculated using a modified power curve, which gave the same surface heat flux on the heater rod as that in the FIST and BWR/6 cores. In the similarity analysis for FIST (case FIST-S), the initial water level in the downcomer was raised so that the initial mass inventory per full-size bundle was the same as in BWR/6 system.

The BWR/6 system was modeled by 69 volumes, 79 junctions and 31 heat slabs, which are similar to those of ROSA-III and FIST. The 848 nuclear fuel bundles were modeled by the central core (748 bundles, component 40) and the peripheral core (100 bundles, component 45). Each component was vertically divided into 8 nodes with 1 non-heated node (top of bundles). Three types of nuclear fuel rods were considered, i.e., the average power rod with local and radial peaking factors of 1.0 × 1.0, the highest power rod with 1.13 × 1.4 located in the central core, and peripheral bundle rods with 1.0 × 0.7. The power curve simulated the ANS decay heat curve. The pump characteristics were modeled by those of the Bingham pump. The PV wall was modeled as heat slabs as in the ROSA-III and FIST analyses.

20.4.2 Similarity of 100% Steam Line Break LOCA Tests at ROSA-III and FIST to BWR/6 LOCA

Figure 20.8 shows the pressure response and the actuation times of HPCS, LPCS and LPCI in the three systems. Differences in the pressure responses between the test results of ROSA-III and FIST (see Fig. 20.3) were almost eliminated in the similarity analyses by applying the same MSIV trip logic, initial mass inventory per full-size bundle and ECC water injections.

The BWR pressure response, however, is slightly different from ROSA-III and FIST due to the following reason. The BWR pressure increased immediately after the break due to smaller volumetric break flow rate because the BWR main steam line was covered by the mixture level earlier than the other two cases. The overall response of the BWR pressure agreed well with those of ROSA-III and FIST resulting in similar actuation times of the LPCS and LPCI. The similar actuation times of the LPCS and LPCI are very important for similarity in the core cooling phenomena which is shown later. The modified core power used in the ROSA-M case affected the pressure response slightly, and resulted in an earlier LPCS actuation (less than 5 s) compared to the ROSA-S case.

Figure 20.9 shows the mass inventory and ECCS water injected in each system per full-size bundle. The transient mass inventory in each case shows a similar decrease after break initiation and a recovery after LPCS actuation. The minimum fluid mass remaining in the system was more than 60% of the initial mass in all cases. The earlier mass recovery
in the BWR case is due to the earlier ECCS actuation, which is a result of faster depressurization caused by the smaller heat release from the vessel wall compared with other two cases. The ROSA-III vessel wall is slightly thicker than the FIST wall, resulting in slower depressurization, smaller amount of ECC water injected and slower recovery of mass inventory. The effect of the relative steam line height in the steam dome affected the decrease of mass inventory after the break.

It is shown in Table 20.5 that (1) the difference in the transient fluid energy between the three systems is small (less than 8%), (2) the released heat from the fuel rods and the discharged fluid energy dominated the energy balance in the early phase of the three systems, and (3) the effects of stored heat in the PV walls of ROSA-III and FIST became larger in the later phase of LOCA (at 100 s after the break and later), whereas the effect of the energy release from the BWR PV walls was negligibly small. The larger amount of heat released from the metal caused more energy to be discharged through the break. The thick PV wall in ROSA-S and BWR resulted in slower heat release and a smaller decrease of metal stored energy in comparison with the FIST test.

Figure 20.10 shows the surface temperatures at the top, middle and bottom of the average power rod in ROSA-S, ROSA-M, FIST-S and BWR. In each case, the fuel rod showed a slight dryout and temperature rise at the top and middle elevations of the core, which was well cooled by actuation of the LPCS and LPCI. The temperature excursion in the ROSA-S case was suppressed by modifying the core power curve (case ROSA-M). Consequently, the temperature responses in the ROSA-M, FIST-S and BWR cases showed good agreement by assuming the same trip logic for the MSIV and ECCS, the same volumetrically scaled mass and energy inventory for the initial conditions, and similar core power in the three different systems.

It is concluded through these comparisons that the thermal-hydraulic phenomena are similar in the 100% steam line break LOCA among the ROSA-III, FIST and BWR systems on the same time scale. The volumetric scaling concept adopted in the similarity analyses resulted in good agreement of the pressure response and the timings of major events. The difference in the geometries and metal stored heat among the three systems, however, caused slight differences in the phenomena and the timings of events.

20.5 Conclusions

The 100% steam line break LOCA was studied experimentally in the ROSA-III and FIST systems. Major conclusions obtained from the tests are as follows.

(1) Similar thermal-hydraulic phenomena were observed in the ROSA-III and FIST tests with respect to the BWR large steam line break LOCA. These phenomena include rapid depressurization in the early blowdown phase, slow depressurization in the later stage due to flashing of a large liquid mass remaining in the system. Also the downcomer mixture level swelled up to the main steam line level.

(2) The time period that the main steam line is covered by the two-phase mixture depends on the geometry of the system and the initial mass inventory.

(3) The heater rod temperature responses were different between the ROSA-III and FIST tests due to different ECCS actuation logics. However, the test results made it possible to investigate separately the effects of each ECCS on the core cooling phenomena. The early actuation of HPCS contributed to prevent the early dryout of the core and actuation of LPCS and LPCI contributed to long term stable core cooling. The similarity of fundamental thermal-hydraulic phenomena was studied analytically.
among these two systems and a BWR/6 with 848 bundles, and following conclusion was obtained.

1) The use of volumetrically scaled initial mass, core power, break flow areas and ECCS injection flow rates, and the same enthalpy distribution in the primary system, resulted in similar occurrence of major events and similar core cooling phenomena among the BWR/6, ROSA-III and FIST systems.

References


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Table 20.5 Balance of fluid energy per full-size bundle
### Table 20.1 Comparison of metal heat source in each region of ROSA-III and FIST

<table>
<thead>
<tr>
<th>Region</th>
<th>1/2 X ROSA-III</th>
<th>FIST</th>
<th>Ratio (ROSA-III/2FIST)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Surface Area (m²)</td>
<td>V (m³)</td>
<td>Surface Area (m²)</td>
</tr>
<tr>
<td>Lower Plenum</td>
<td>6.9</td>
<td>0.195</td>
<td>5.7</td>
</tr>
<tr>
<td>Guide Tube</td>
<td>1.6</td>
<td>0.010</td>
<td>1.5</td>
</tr>
<tr>
<td>Core Bypass</td>
<td>5.2</td>
<td>0.015</td>
<td>5.1</td>
</tr>
<tr>
<td>Core</td>
<td>15.0</td>
<td>0.043</td>
<td>12.6</td>
</tr>
<tr>
<td>Upper Plenum</td>
<td>0.7</td>
<td>0.005</td>
<td>1.0</td>
</tr>
<tr>
<td>Separator</td>
<td>1.0</td>
<td>0.002</td>
<td>2.4</td>
</tr>
<tr>
<td>Downcomer</td>
<td>5.5</td>
<td>0.240</td>
<td>8.8</td>
</tr>
<tr>
<td>Steam Dome</td>
<td>2.9</td>
<td>0.140</td>
<td>5.1</td>
</tr>
<tr>
<td>Loops &amp; JPs</td>
<td>5.6</td>
<td>0.095</td>
<td>1.2</td>
</tr>
<tr>
<td>Total</td>
<td>44.5</td>
<td>0.745</td>
<td>43.4</td>
</tr>
<tr>
<td>Average Thickness</td>
<td>16.9 mm</td>
<td>9.7 mm</td>
<td></td>
</tr>
</tbody>
</table>

### Table 20.2 Test conditions of ROSA-III (Run 952) and FIST (6MSB1)

<table>
<thead>
<tr>
<th>Items</th>
<th>Units</th>
<th>ROSA-III</th>
<th>FIST</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Initial Condition</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Steam Dome Pressure</td>
<td>(MPa)</td>
<td>7.35</td>
<td>7.17</td>
</tr>
<tr>
<td>Core Power</td>
<td>(MW)</td>
<td>1.98 *</td>
<td>4.62</td>
</tr>
<tr>
<td>Upper Plenum Quality</td>
<td>(%)</td>
<td>13.0</td>
<td>16.0</td>
</tr>
<tr>
<td>Total Core Flow Rate</td>
<td>(kg/s)</td>
<td>8.3 *</td>
<td>17.0</td>
</tr>
<tr>
<td>Total Mass Inventory**</td>
<td>(kg)</td>
<td>354 *</td>
<td>287</td>
</tr>
<tr>
<td><strong>Break Condition</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Break Location</td>
<td></td>
<td>Main Steam Line</td>
<td>Main Steam Line</td>
</tr>
<tr>
<td>Break Area</td>
<td>(mm²)</td>
<td>377 *</td>
<td>501/377</td>
</tr>
<tr>
<td><strong>Transient Condition</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Power Curve Coastdown</td>
<td>s</td>
<td>9 s after Break</td>
<td>at Break Time</td>
</tr>
<tr>
<td>Trip of Recirculation Pump</td>
<td>s</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>s</td>
<td>0.0</td>
<td>5.5</td>
</tr>
<tr>
<td>Feedwater Termination</td>
<td>s</td>
<td>1.3 – 3.2</td>
<td>0.0</td>
</tr>
<tr>
<td>HPCS Actuation</td>
<td>s</td>
<td>1.2 + 27</td>
<td>27.0</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>(s)</td>
<td>1.1 + 40</td>
<td>35</td>
</tr>
<tr>
<td></td>
<td>(MPa)</td>
<td>P ≤ 2.2</td>
<td>P ≤ 2.0</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>(s)</td>
<td>1.1 + 40</td>
<td>35</td>
</tr>
<tr>
<td></td>
<td>(MPa)</td>
<td>P ≤ 1.6</td>
<td>P ≤ 1.7</td>
</tr>
<tr>
<td><strong>Assumption of Failure</strong></td>
<td></td>
<td>Signal of High Containment Press, 2-LPCI</td>
<td></td>
</tr>
</tbody>
</table>

* Volumetrically scaled value (half value in ROSA-III test).
** Calculation results of the post-test analysis including steam mass. The data of ROSA-III does not include feedwater line volume.
Table 20.3  Comparison of major events in ROSA-III and FIST tests

<table>
<thead>
<tr>
<th>Key Events</th>
<th>ROSA-III</th>
<th>FIST</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break Initiation</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>Pump Coastdown</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>Feedwater Stop</td>
<td>1.3–3.2</td>
<td>0.0</td>
</tr>
<tr>
<td>Power Decay Initiation</td>
<td>9.0</td>
<td>0.0</td>
</tr>
<tr>
<td>Lower Plenum Flashing</td>
<td>4.2</td>
<td>5.0</td>
</tr>
<tr>
<td>MSIV Closure</td>
<td>0.0</td>
<td>5.5</td>
</tr>
<tr>
<td>Level Covers Steamline</td>
<td>15.</td>
<td>6.</td>
</tr>
<tr>
<td>Level 2 Reached</td>
<td>71.</td>
<td>22.</td>
</tr>
<tr>
<td>Level 1 Reached</td>
<td>No</td>
<td>No</td>
</tr>
<tr>
<td>Level Uncovers Steamline</td>
<td>31.</td>
<td>80.</td>
</tr>
<tr>
<td>HPCS Actuation</td>
<td>94.</td>
<td>27.</td>
</tr>
<tr>
<td>LPCS Actuation</td>
<td>No</td>
<td>88.</td>
</tr>
<tr>
<td>LPCI Actuation</td>
<td>No</td>
<td>95.</td>
</tr>
<tr>
<td>Feedwater Line Flashing</td>
<td>95.</td>
<td>–</td>
</tr>
<tr>
<td>Core Heatup Begins</td>
<td>27.</td>
<td>No</td>
</tr>
<tr>
<td>Final Rod Quench</td>
<td>221.</td>
<td>–</td>
</tr>
<tr>
<td>PCT Elevation</td>
<td>50% Height</td>
<td>High P. Rod</td>
</tr>
</tbody>
</table>

### Table 20.4 Analytical conditions for post-test and similarity analyses

#### (a) Analytical Conditions of FIST Test (6MSB1)

<table>
<thead>
<tr>
<th>Items</th>
<th>Models and Analysis Conditions</th>
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<tbody>
<tr>
<td>Analysis Code</td>
<td>RELAP5/Mod1/CY18</td>
</tr>
<tr>
<td>Analysis Models</td>
<td>Non-Homogeneous, Separator, Jet Pump (JAERI), Dis. Coef. (C_d=0.5)</td>
</tr>
<tr>
<td>Components</td>
<td>56 Vols, 63 Juns, 27 Heat Slabs</td>
</tr>
<tr>
<td>Noding of Core</td>
<td>1 Pipe (8 axial volumes)</td>
</tr>
<tr>
<td>Initial/Break Conditions</td>
<td>Test Conditions of 6MSB1</td>
</tr>
<tr>
<td>Steam/Feedwater Systems</td>
<td>Test Conditions of 6MSB1</td>
</tr>
<tr>
<td>ECCS/Core Power Conditions</td>
<td>Test Conditions of 6MSB1</td>
</tr>
</tbody>
</table>

#### (b) Analytical Conditions of ROSA-III Test (Run 952)

<table>
<thead>
<tr>
<th>Items</th>
<th>Models and Analysis Conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Analysis Code</td>
<td>RELAP5/Mod1/CY18</td>
</tr>
<tr>
<td>Analysis Models</td>
<td>Non-Homogeneous, Separator, Pump, Jet Pump (JAERI), Dis. Coef. (C_d=0.5)</td>
</tr>
<tr>
<td>Components</td>
<td>63 Vols, 66 Juns, 26 Heat Slabs</td>
</tr>
<tr>
<td>Noding of Core</td>
<td>2 Pipes (7 axial volumes)</td>
</tr>
<tr>
<td>Initial/Break Conditions</td>
<td>Test Conditions of RUN 952</td>
</tr>
<tr>
<td>Steam/Feedwater Systems</td>
<td>Test Conditions of RUN 952</td>
</tr>
<tr>
<td>ECCS/Core Power Conditions</td>
<td>Test Conditions of RUN 952</td>
</tr>
</tbody>
</table>

#### (c) Analytical conditions of ROSA-III, FIST and BWR/6 for similarity elevation

<table>
<thead>
<tr>
<th>Items</th>
<th>ROSA-III</th>
<th>FIST</th>
<th>BWR/6 (251)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Scaling Ratios</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A. Total Fluid Volume Ratio **</td>
<td>1.82</td>
<td>0.93</td>
<td>848</td>
</tr>
<tr>
<td>B. Initial Mass Inventory Ratio **</td>
<td>2.00 *</td>
<td>1.00</td>
<td>848</td>
</tr>
<tr>
<td>C. Break Area Ratio</td>
<td>2.00</td>
<td>1.00</td>
<td>848</td>
</tr>
<tr>
<td>D. ECCS Flow Rate Ratio</td>
<td>2.00 *</td>
<td>1.00</td>
<td>848</td>
</tr>
<tr>
<td>E. Initial Core Power Ratio</td>
<td>0.44</td>
<td>1.03</td>
<td>848</td>
</tr>
<tr>
<td>F. Initial Core Flow Ratio</td>
<td>0.44</td>
<td>0.93</td>
<td>848</td>
</tr>
<tr>
<td>G. Enthalpy Distribution (E)/(F)</td>
<td>1.01</td>
<td>1.11</td>
<td>1.00</td>
</tr>
<tr>
<td>H. PV Wall Surface Area Ratio</td>
<td>21.9</td>
<td>34.0</td>
<td>848</td>
</tr>
<tr>
<td>I. PV Wall Metal Volume Ratio</td>
<td>12.0</td>
<td>4.24</td>
<td>848</td>
</tr>
<tr>
<td>J. Fluid-Contact Area Ratio (H)/(A)</td>
<td>11.9</td>
<td>36.6</td>
<td>1.00</td>
</tr>
</tbody>
</table>

| Transient Conditions               |          |      |             |
| K. MSIV Closure Time (s)           | 0.0–5.5 *| 0.0–5.5 | 0.0–5.5 |
| L. Feedwater Line Closure Time (s) | 2.0–3.1 *| 2.0–3.1 | 2.0–3.1 |
| M. HPCS Actuation Time (s)         | 27 *     | 27   | 27          |
| N. LPCS Actuation Pressure (MPa)   | 2.1      | 2.1  | 2.1         |
| O. LPCI Actuation Pressure (MPa)   | 1.55     | 1.55 | 1.55        |

<table>
<thead>
<tr>
<th>P. Core Power Curve/(Case)</th>
<th>Post-Test Analysis (ROSA-S)</th>
<th>Post-Test Analysis (FIST-S)</th>
<th>ANS Curve (BWR)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Modified Power (ROSA-M)</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Altered from the post-test analysis conditions
** Include steam mass and exclude feedwater mass
<table>
<thead>
<tr>
<th>Case</th>
<th>Items</th>
<th>Unit</th>
<th>Time after Break (s)</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>0</td>
<td>35</td>
</tr>
<tr>
<td>ROSA-S</td>
<td>A. Released Heat from PV Wall</td>
<td>$10^3$kJ</td>
<td>0.03</td>
<td>0.19</td>
</tr>
<tr>
<td></td>
<td>B. Release Heat in Core</td>
<td>&quot;</td>
<td>0.44</td>
<td>0.66</td>
</tr>
<tr>
<td></td>
<td>C. Discharged Energy at Break</td>
<td>&quot;</td>
<td>-1.64</td>
<td>-3.17</td>
</tr>
<tr>
<td></td>
<td>D. Injected Water Energy</td>
<td>&quot;</td>
<td>0.01</td>
<td>0.07</td>
</tr>
<tr>
<td></td>
<td>E. Remained Fluid Energy</td>
<td>&quot;</td>
<td>4.22</td>
<td>3.06</td>
</tr>
<tr>
<td>FIST-S</td>
<td>A. Released Heat from PV Wall</td>
<td>&quot;</td>
<td>0.05</td>
<td>0.35</td>
</tr>
<tr>
<td></td>
<td>B. Released Heat in Core</td>
<td>&quot;</td>
<td>0.37</td>
<td>0.51</td>
</tr>
<tr>
<td></td>
<td>C. Discharged Energy at Break</td>
<td>&quot;</td>
<td>-1.74</td>
<td>-3.34</td>
</tr>
<tr>
<td></td>
<td>D. Injected Water Energy</td>
<td>&quot;</td>
<td>0.01</td>
<td>0.08</td>
</tr>
<tr>
<td></td>
<td>E. Remained Fluid Energy</td>
<td>&quot;</td>
<td>4.24</td>
<td>2.93</td>
</tr>
<tr>
<td>BWR</td>
<td>A. Released Heat from PV Wall</td>
<td>&quot;</td>
<td>0.01</td>
<td>0.03</td>
</tr>
<tr>
<td></td>
<td>B. Released Heat in Core</td>
<td>&quot;</td>
<td>0.46</td>
<td>0.59</td>
</tr>
<tr>
<td></td>
<td>C. Discharged Energy at Break</td>
<td>&quot;</td>
<td>-1.70</td>
<td>-2.97</td>
</tr>
<tr>
<td></td>
<td>D. Injected Water Energy</td>
<td>&quot;</td>
<td>0.01</td>
<td>0.08</td>
</tr>
<tr>
<td></td>
<td>E. Remained Fluid Energy</td>
<td>&quot;</td>
<td>4.27</td>
<td>3.05</td>
</tr>
</tbody>
</table>
Fig. 20.1  Configuration of test facilities of ROSA-III and FIST.

Fig. 20.2  Comparison of transient power curves between ROSA-III and FIST.
Fig. 20.3  Comparison of test data of pressure.

Fig. 20.4  Comparison of test data of mixture levels in downcomer.
Fig. 20.5 Comparison of test data of rod surface temperature.
Fig. 20.6  Nodalization of ROSA-III system.
Fig. 20.7  Nodalization of FIST system.
Fig. 20.8 Pressure responses and major events in similarity analyses of ROSA-III, FIST and BWR/6.

Fig. 20.9 Total mass inventory and injected ECC water per full-size bundle in similarity analyses of ROSA-III, FIST and BWR/6.
Fig. 20.10 Surface temperatures of average-power rod in similarity analyses of ROSA-III, FIST and BWR/6.
21. Similarity between ROSA-III and BWR Thermal-Hydraulic Responses

21.1 Introduction

Several earlier Chapters of this report have investigated the applicability of the ROSA-III experimental results to actual BWRs. These Chapters concluded that, for representative LOCA transients, good qualitative similarity can be expected between the thermal-hydraulic responses of ROSA-III and the reference BWR. Quantitatively, however, differences are found between the detailed responses of the two systems, since the ROSA-III responses are influenced by the facility scaling and scaling distortions.

These earlier Chapters evaluated the similarity between the responses of the two systems by using several computer codes as summarized in Table 21.1. This table also includes analyses which have not been referred to in the earlier Chapters. In all of these analyses the similarity was examined by comparing the ROSA-III test results with code-predicted BWR responses to equivalent boundary conditions (break conditions, ECCS failure mode, etc.), or, more often, by comparing the code-predicted response of ROSA-III to that of the reference BWR. The codes used for this purpose, i.e., THYDE-BI MOD1 and MOD2, RELAP5/MOD1 CYs 01 and 183 (JAERI versions with a jet pump model) and RELAP4/MOD6/U4/J33, were assessed against ROSA-III data for accuracy, before making such comparisons. The input modeling and code input parameters were assessed as well. These model and parameters were adjusted, when necessary, so that reasonable agreement was obtained between the predicted and measured ROSA-III responses.

The accuracy of this approach for the evaluation of similarity between tests and plant responses is dependent, of course, on computer code's ability in predicting the plant responses. Accordingly, the present work used those computer codes which were built with careful consideration for application to reactor accident analysis. These codes may predict the plant responses with reasonable accuracy, if they have been well assessed and verified against test data, on conditions that the tests closely represent the plant responses. It is the present authors' understanding that the ROSA-III test simulated most of the controlling phenomena in BWR LOCAs under conditions not too far from the actual accidental conditions. The similarity between the ROSA-III and actual BWR responses was studied also experimentally, by conducting counterpart tests using the ROSA-III and FIST facilities. These counterpart tests were useful to understand the scaling effects on thermal-hydraulic responses since these two facilities took different scaling approaches. Thus these tests supplement the similarity studies conducted with use of computer codes.

This Chapter summarizes the conclusions obtained from these similarity analyses. It also investigates several specific problems related to the applicability of the ROSA-III test results to actual BWRs, which have not been discussed in the earlier Chapters explicitly. The following discussions apply to the final ROSA-III facility geometry (during the Series 900 tests), the standard test conditions and the standard test procedures.

21.2 Scaling Criteria

The ROSA-III facility and experiments were designed to simulate, on a real-time basis, full-scale pressure and temperature transients during a BWR LOCA, using the same working
fluid (water) as that of an actual BWR. The facility overall scaling factors are 1/424 in volume, and 1/2 in height. Such scaling criteria were led from the design of the simulated core, which represents four half-length full-diameter (full-scale channel box cross-sectional dimensions) bundles. Accordingly, the major parameters and their desired scales are:

<table>
<thead>
<tr>
<th>parameter</th>
<th>scale</th>
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</thead>
<tbody>
<tr>
<td>time</td>
<td>1/1</td>
</tr>
<tr>
<td>pressure</td>
<td>1/1</td>
</tr>
<tr>
<td>temperature</td>
<td>1/1</td>
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<tr>
<td>level</td>
<td>1/2</td>
</tr>
<tr>
<td>differential pressure</td>
<td>1/2</td>
</tr>
<tr>
<td>mass, volume</td>
<td>1/424</td>
</tr>
<tr>
<td>flow rate</td>
<td>1/424</td>
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<tr>
<td>break area(s)</td>
<td>1/424</td>
</tr>
<tr>
<td>core power</td>
<td>1/424</td>
</tr>
<tr>
<td>core heat flux</td>
<td>1/1</td>
</tr>
</tbody>
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The 1/424-volume scaling and the 1/2-height scaling, combined together, require horizontal cross-sectional areas to be scaled by 1/212. Then, the desired scale for fluid vertical velocity is 1/2, since the flow rates are to be scaled volumetrically (1/424 scale). However, full-scale (1/1) velocity scaling (or 1/424 flow area scaling, in other words) was also considered for those portions with flow area restriction that may control the counter-current flow limiting (CCFL) behavior, or dynamic pressure drop, or critical discharge flow rate. Such portions include the core inlet orifice, the lower and upper tieplates (LTP and UTP), the jet pump drive nozzle and mixing region, and the simulated break.

The selection of these scaling criteria led to radial dimensions much smaller than in actual BWRs. Consequently, most of the flow paths are of one-dimensional geometries. This is not serious because most of the flow paths in an actual BWR are also one-dimensional. In particular, the core geometry of a BWR, divided into channels each one containing a small number (about 60) of fuel rods, is suitable for experimental representation using a limited number of rods. As the radial dimension of ROSA-III is smaller than that of a BWR, ROSA-III's ability is limited in representing such two- or three-dimensional fluid motions as are extending over the full radius of the pressure vessel. Such multidimensional phenomena may include the parallel channel behavior of the core bundles, the parallel channel behavior of the jet pumps, and the nonhomogeneous water temperature distribution in the upper plenum.

### 21.3 Scaling Distortions

The above scaling criteria are not met in all the areas of the facility design. The scaling distortions arose from four major sources, which are: (i) 1/424-volume scaling, (ii) 1/2-height scaling, (iii) core power limitation and atypical core material, and (iv) distorted (or simplified) component geometries. Important scaling distortions are summarized and discussed in the following paragraphs.

The 1/424-volume scaling causes excessive stored heat release in ROSA-III experiments. In a subscale system, it is difficult to scale correctly the structure stored heat, since the structure surface areas become necessarily greater than the volumetrically-scaled values. Hence the stored heat release is greater than scaled, unless special care (e.g., use of thermally insulating materials for facility internal surfaces) is taken in the facility design.
The stored heat effect is significant during the depressurization phase of a LOCA experiment where fluid saturation temperature decreases with time, and during the refilling phase where structure wetted area increases with time. The effect is important for those regions where liquid is in contact with massive structures (e.g., vessel wall). Such portions in ROSA-III include the lower plenum and the downcomer.

The 1/2-height scaling leads to inconsistency between the gravitational head and dynamic pressure loss in several portions of the facility, where pressure losses for volumetrically-scaled flow rates are larger than 1/2 scale (see Section 21.2). However, this inconsistency between the dynamic pressure loss and the gravitational head is unimportant except for limited situations mainly during the early phase of a large break transient, where fluid velocities can be large. For most of the BWR LOCA transients, the fluid motions are controlled primarily by the gravitational head differences.

The ROSA-III steady-state core power is about 40% (typically between 42 and 44%) of the volumetrically-scaled BWR rated core power, because of power supply limitations. The ROSA-III steady state is designed to simulate the primary coolant enthalpy distribution during BWR rated operating conditions within this limitation of core power. To do this, power-scaled flow rates are established in the system. Namely, the steady-state core flow rate, main steam flow rate and feedwater flow rate are controlled at about 40% of the volumetrically-scaled BWR flow rates. By this means, the ROSA-III core inlet temperature and core exit quality closely represent the BWR nominal conditions. However, the ROSA-III steady-state bundle void fraction is slightly (about 10%) lower than that in the reference BWR. Fig. 21.1 compares the steady-state bundle void fraction distribution in ROSA-III and the reference BWR. The void fractions were calculated for average bundle power and average bundle flow rate in each system. A drift flux model applicable to rod bundle geometry was used for the calculation. The calculated void fractions are lower for ROSA-III than for the reference BWR, although the core exit quality is the same for the two systems, because of a smaller bundle vapor volumetric flux in ROSA-III than in the reference BWR.

ROSA-III simulates the BWR nuclear fuel rods using electric heater rods. The rod steady-state stored heat (per unit length) is smaller than that of a BWR fuel rod, because of higher thermal conductivity of rod material and a lower steady-state core power (per unit length). This difference is taken into consideration in developing the ROSA-III core power decay curve.

The ROSA-III core power decay curve was designed to simulate the time-dependent rod-surface heat transfer rate, rather than the core power generation rate, of a BWR following scram. The smaller steady-state rod stored heat is compensated by providing larger core power than the volumetrically scaled BWR core power, during the initial period following scram. The ROSA-III core power decay curve is determined conservatively, except for the first 9 s after scram where the rod surface heat transfer rate (per unit length) is smaller than that in the reference BWR because of the aforementioned power supply limitations.

The ROSA-III regional cross-sectional areas are not exactly scaled at 1/212 for all the portions of the facility. If this is done, many portions will have pencil-like configurations and will suffer from excessive stored heat release and atypical two-phase flow behavior. Thus, there are several portions in the facility where the elevation and cross-sectional area are not correctly scaled, whereas the volumes are scaled. Such portions include the steam dome, the downcomer, the lower plenum and the recirculation lines.
21.4 Overall Responses of ROSA-III and Reference BWR

This Section describes briefly the overall responses of ROSA-III and the reference BWR for representative test/accident conditions, before discussing the responses of specific parameters in the two systems.

A BWR is a direct-cycle, forced-circulation nuclear steam supply system (NSSS). According to the results from the ROSA-III experiments, the design features characteristic to a BWR leads to the following thermal-hydraulic responses peculiar to a BWR LOCA.

(i) Early and rapid system depressurization

It is characteristic to a BWR that a major rupture leads to a system depressurization automatically, irrespective of the size of rupture, before significant amount of primary mass inventory is lost. Early system depressurization is important to core coolability during a LOCA, since it enables ECCS to deliver sufficient amount of coolant into the system. The efficient way of depressurizing a two-phase system, without losing a large amount of mass, is to discharge vapor from the top region of the system. A BWR LOCA leads to major vapor discharge automatically. For instance, a large break in the maximum-size liquid line, the recirculation line, empties the downcomer quickly, and thus leads to vapor discharge through the uncovered nozzle of the broken recirculation line. Smaller breaks lead to the operation of ADS, tripped on by the downcomer level drop. ADS releases vapor through valves on the main steam line, with a total flow area corresponding to about 60% of the recirculation line flow area, and thus depressurizes the system efficiently.

(ii) Gravitation-controlled mass redistribution

The BWR vessel internal geometry is designed to limit the depletion of core mass inventory during a LOCA. The vessel internal volume is divided by the core shroud into the downcomer and core regions. The two regions communicate with each other at the top of core region through the separator, and at the bottom of downcomer through the jet pumps. The jet pump suction opens in the downcomer, at an elevation corresponding to about 2/3 height of the core. This configuration limits the decrease of core level following a rupture of the recirculation line connected to the downcomer. Because of such a vessel internal geometry, the redistribution of the vessel mass inventory during a LOCA is controlled essentially by the gravitational head difference between the downcomer and core regions, as long as the jet pump suction is submerged.

(iii) Early system saturation

Fluid is saturated or nearly saturated all through the primary system during the normal operation of a BWR. The core inlet subcooling is about 10 K. Thus, depressurization following a LOCA leads quickly to saturation, flashing and boiling in the whole primary system. The flashing and boiling in the lower portion of the pressure vessel, i.e., the lower plenum, have an important influence on the system response, since it causes mixture level swell in the core, CCFL at the core inlet, and vapor updraft through the core. All of these influence favorably the core cooling, by holding up the core mixture level, delaying the core emptying, and causing the core to be reflooded rapidly once the LPCI water reaches the core inlet region.

In addition, relatively low heat ratings of fuel rods, and the ECC injection locations close to the core (particularly for BWR/6) contribute to the safety of a BWR during a LOCA.

These design features and characteristic response of BWR have been well taken into consideration in designing the ROSA-III facility and experiments, as well as in analyzing the experimental results.

Figures 21.1 and 21.2 compare the vessel pressures, core mixture levels and rod surface temperatures (at the core midplane) in ROSA-III and the reference BWR, for typical small
and large breaks (5% and 200% in area) in the main recirculation pump suction line, followed by the failure of HPCS, respectively. These responses were calculated\textsuperscript{7,8} using RELAP4/MOD6/U4/J3\textsuperscript{3} which is a JAERI-improved version of the RELAP4/MOD6 code. The good agreement between the predicted ROSA-III responses with the experimental data shows that the code has sufficient ability for the present purpose of discussing the similarity between the ROSA-III and reference BWR overall responses.

First of all, qualitative similarity between the small-break and large-break transients is notable, as has been discussed in the earlier Chapters, e.g., Chapter 5. For both cases, and for both systems, a rapid depressurization occurs. The depressurization is accompanied by a decrease in the core mixture level, which causes the core to uncover. Then, a temperature excursion occurs on the uncovered rod surfaces until the emergency core coolant (ECC) rewets the rod surfaces. For a small break, e.g., a 5% break, the system response is dictated by the operation of ADS, which causes a rapid system depressurization as well as a loss of primary mass inventory. The system response for a large break, e.g., a 200% break, is dictated by the flow out of the break, since ADS is tripped on only after the vessel has depressurized. The good overall agreement between the ROSA-III and reference BWR pressures results primarily from good simulation by ROSA-III of flow rates through the break and ADS.

The core mixture level drop results from the loss of mass inventory as well as the mass redistribution. The ROSA-III mixture level agrees well the 1/2-scaled BWR level, since ROSA-III simulates well the geometries affecting the gravitation-controlled mass redistribution.

The rod temperature closely responds to the core mixture level. Thus, ROSA-III simulates the overall response of BWR rod surface temperature.

The following Sections will discuss more in detail the similarity between the ROSA-III and reference BWR responses in terms of vessel pressure, core mixture level and rod temperatures.

21.5 Similarity between ROSA-III and BWR Vessel Pressure Responses

The ROSA-III experiments were designed to represent the full-scale pressure response of the reference BWR isochronously. The simulation of vessel pressure response is important, because it controls the vessel mixture level swell, ECCS timings and flow rates, and thus core rod temperature.

The vessel pressure transient during a LOCA is dictated by the vapor discharge timing and flow rate out of the vessel, as well as vapor generation rate in the vessel. This is clearly seen in Fig. 21.3 where vessel pressure responses are shown for recirculation-pump suction-line break tests with different break sizes\textsuperscript{12}. The break flow is initially liquid for these tests. Then, the vessel depressurizes only slowly until vapor discharge initiates. The timing of the initiation of vapor discharge is dependent much on the size of break. For a large break, vapor discharge starts as the downcomer level drops to uncover the recirculation pump suction-line nozzle. For a small break, however, vapor discharge through ADS occurs earlier than the recirculation line recovery (RLU). The fluid in the downcomer remains subcooled until the initiation of vapor discharge, because the vessel pressure is kept above the downcomer saturation pressure (~6.4 MPa). The similarity between the BWR and ROSA-III pressure transients is discussed for three representative cases, which are: (i) a small break in the recirculation-pump suction line, (ii) a large break at the same location, and (iii) a main steam line break.
21.5.1 Small Break in Recirculation-Pump Suction Line

Three periods are identified in the pressure transient caused by a small recirculation-line break (see, e.g., Fig. 21.1): (i) an almost-constant pressure phase which continue until the initiation of ADS, (ii) a rapid depressurization caused by ADS, and (iii) a pressure halt and subsequent depressurization following the initiation of the low pressure ECCSs (LPCS and LPC1).

The first phase, more in detail, includes a short-term depressurization and a subsequent repressurization reaching the safety relief valve (SRV) setpoint. The first depressurization, caused by the break flow, is arrested as the pressure control system throttles the main steam valve. Then the closure of MSIV, tripped by a low downcomer level causes the vessel to repressurize. The ROSA-III pressure control logic simulates that of a BWR, however, the setpoint pressure (6.7 MPa) is referred at different locations (pressure vessel in ROSA-III vs. turbine inlet in the reference BWR, see Chapter 8). The MSIV closure, tripped by a Level-2 signal with a 3-s delay, occurs later in ROSA-III than that predicted for the reference BWR, since: (i) the ROSA-III steady-state downcomer water mass above Level 2 is greater than scaled, and (ii) the ROSA-III downcomer level deops more slowly than that in the reference BWR.

The downcomer level drop in this period is affected by, in addition to the break flow, the coasting-down jet pump flow from the downcomer to the lower plenum, and the bundle void collapse due to core power decay. The ROSA-III jet pump flow rate during this period is smaller than the volumetrically-scaled BWR flow rate, because of a smaller-than-scaled steady-state flow rate and a faster cooldown of the recirculation pump speed. The bundle void collapse increases the downcomer level drop in two ways: (i) it increases the jet pump flow, since the void collapse decreases the bundle dynamic pressure drop; and (ii) decreases the liquid recirculation flow from the separator to the downcomer. These two effects of void collapse are less significant in ROSA-III than in the reference BWR, because of: (a) a delayed core power decay (the steady-state core power is maintained during the first 9 seconds after break); (b) a lower steady-state bundle void fraction (Fig. 21.4); and (c) a lower setpoint pressure of the pressure control system which allows more vessel depressurization before the MSIV closure. Also, the earlier MSIV closure in BWR amplifies the void collapse effects, since the MSIV closure causes a vessel repressurization and further void collapse. These differences combined, cause slower downcomer level decrease in ROSA-III than in the reference BWR.

The vessel repressurization following the MSIV closure opens the SRV(s). The ROSA-III SRV setpoint pressure simulates the intermediate setpoint of the three groups of the reference BWR SRVs. Thus, both ROSA-III and BWR pressures are kept at similar values after the opening of SRV(s). Then, the differences between the ROSA-III and BWR responses which exist before the opening of SRV, caused primarily by the smaller-than-scaled ROSA-III steady-state core power, have only small influence on the later system responses.

The ADS is tripped on by a downcomer Level-1 signal with a 120-s time delay, and opens relief valve(s) on the main steam line to cause a rapid depressurization of the primary system. For a very-small break LOCA, Level 1 is reached a few hundreds of seconds after break since the loss of downcomer fluid out of the break is slow. Such a slow decrease of the downcomer level is influenced strongly by the manometric balance between the fluid gravitational heads in the downcomer and in-shroud regions. Thus, correct height scaling of downcomer level setpoints (Levels 1 through 3) is desirable. However, the ROSA-III downcomer level setpoints are determined so that downcomer fluid volumes below these levels are scaled (Fig. 21.5), but are not height-scaled, since the downcomer flow area (versus height) is not exactly scaled. As a result, the ROSA-III Level-1 setpoint (relative to the bundle elevation) is higher than the height-scaled BWR setpoint. This causes earlier ADS initiation in
ROSA-III than in the reference BWR for a very small break, e.g., 1% or less (see Fig. 7.6).

The ROSA-III ADS flow capacity is 60% of the volumetrically-scaled BWR capacity. This ratio corresponds to the ADS orifice flow coefficient in ROSA-III. The smaller-than-scaled ADS flow capacity affects the vessel depressurization rate, mixture level swell, and rod temperature after the initiation of ADS. In the earlier Chapters the ROSA-III experimental data were compared with predicted response of the reference BWR with a 60% ADS capacity, to consider separately the effect of this atypical boundary condition.

Fluid in the lower plenum begins to flash when the vessel depressurizes to 6.4 MPa. Flashing occurs at the same time in other initially-subcooled regions, i.e., the core bypass, the (intact) recirculation line and the downcomer. Thereafter, the stored heat effects become important for the lower plenum, where liquid is in contact with massive vessel walls. The stored heat effects in the downcomer depends on the amount of liquid available there.

The vessel pressure response during the depressurization phase is determined by the break flow rate and flow composition (almost pure vapor for both ADS and the uncovered break), and the rate of in-vessel vapor generation due to boiling and flashing. The boiling rate changes as the structure heat transfer area (wetted by liquid or mixture) changes with the decrease or increase of in-vessel mixture level. Therefore, for correct simulation of the pressure response in this phase, both the mixture level swell (particularly in the core) and the structure stored heat release should be correctly simulated. The code-predicted vessel pressure responses of ROSA-III and the reference BWR indicate almost the same depressurization rates (e.g., see Fig. 21.1), when the same ADS flow capacity is assumed, because ROSA-III simulates the core mixture level well, as will be discussed in Section 21.6, and because the stored heat effects are not very significant for the depressurization rates following the ADS actuation in a small-break transient.

At a vessel pressure of \( \sim 2.2 \) MPa the liquid remaining in the feedwater line begins to flash. The feedwater flashing (FWF) drives the two-phase mixture in the feedwater line into the upper downcomer region. Almost at the same pressure, LPCS begins injection into the upper plenum. Both events cause additional vapor generation as the injected liquid boils on hot vessel structures and internals, and slows the vessel depressurization temporarily. Such vessel pressure response following FWF and LPCS initiation is important, since it affects the timing of core reflooding caused by LPCI; the injection from LPCI starts as the vessel depressurizes below the LPCI shutoff head (1.65 MPa). ROSA-III represents typical BWR feedwater temperature and scaled volume of feedwater line, and scaled LPCS flow rates vs. pressure. However, the excessive stored heat in ROSA-III may affect the pressure response following FWF and LPCS. The code analyses have failed to simulate the halt of pressure following FWF and LPCS, because the existing codes have limited capability in predicting non-equilibrium non-homogeneous phenomena like the direct-contact vapor condensation on LPCS water and boiling of LPCS- and FWF.injected water on hot in-vessel structures.

### 21.5.2 Large Break in Recirculation-Pump Suction Line

The downcomer level drop following a large break in the recirculation pump suction line is fast, because of large flow rates out of the break. The break flow rate is affected by the downcomer liquid subcooling, which is well simulated in ROSA-III, as well as the break upstream geometry (pipe size, pipe length and geometry of break itself) which is not exactly simulated in ROSA-III. However, sensitivity experiments conducted using orifice and nozzle to simulate the break have shown that the break upstream geometry makes only small difference in the timings of downcomer level trips and break recovery. For a large break MSIV closes before the vessel pressure drops to the setpoint of the pressure control system.
After the MSIV closure, the vessel pressure increases until the break uncovery.

The predicted BWR pressure response agree very well with ROSA-III data during the initial part of the depressurization phase. However, during the later phase of a transient, the ROSA-III pressure tends to be higher than the predicted BWR pressure, primarily because of excessive stored heat release. For break sizes larger than the ADS flow area, the stored heat effects are more significant than for the smaller breaks.

The vessel internal surfaces are the most important stored heat sources during the depressurization phase. An estimation with use of the RELAP4/Mod6/U4/J3 code for a 200% break experiment\(^9\) shows that the heat release to the lower plenum amounts as large as 1 MW for about 100 s after the initiation of LPF. This means that the stored heat release to this particular region exceeds the core power after 50 s following scram. The overall stored heat release during a large (200%) break test has also been evaluated in Section 13. The evaluation shows a significant contribution of the stored heat release to the in-vessel energy balance.

### 21.5.3 Steam Line Break

For a steam line break, a gross depressurization of the vessel starts immediately after the break. Most of the ROSA-III steam-line break experiments simulated a break upstream of MSIV. Thus the vessel pressure response is determined simply by the vapor discharge rate through the break and the vapor generation rate in the vessel.

The break flow rate is affected, for a large break, temporarily by the vessel mixture level swell when it covers the steam line inlet. However, the break flow is pure steam for most of the transient, and is determined essentially by the vessel pressure and break size alone.

Large depressurization rates are observed for large sizes of break, and hence the stored heat effects are important. Stored heat release should be taken into consideration not only for the lower plenum but also for the downcomer, because the downcomer remains full of two-phase mixture during the depressurization. The stored heat release rate was calculated for a 100% break experiment using the RELAP5/Mod1/Cy01 code\(^13\) showing that the stored heat release becomes of similar magnitudes to the core power after 70 s following scram.

### 21.6 Similarity between ROSA-III and BWR Vessel Mixture Level Responses

Mixture level swell occurs in both ex-shroud (downcomer) and in-shroud regions of the pressure vessel, as voids are formed in the mixture by flashing due to pressure transient, and by boiling due to heat transfer from structures. Of course, the in-shroud mixture level response is of more interest, because it controls the core uncovery and heat-up.

The mixture level is determined by regional mass inventory and void fraction. Since the ROSA-III experiments are designed to scale the time-dependent in-vessel mass inventory, by scaling volumetrically the break area, it is of interest how well the mass distribution and the void fraction distribution are simulated.

During a LOCA transient, redistribution of mass occurs between the downcomer and in-shroud regions. The mass redistribution is essentially gravitation controlled. The two regions communicate at the top through the vapor spaces, and at the bottom through the jet pump exit connected to the lower plenum. Thus, the in-shroud residual mass inventory is strongly affected by the manometric balance between the fluid gravitational heads in the downcomer and in-shroud regions.

The void distribution is affected by the scaling of two geometrical parameters: height
and characteristic diameter. The ROSA-III core represents full-scale characteristic diameters (channel box diameter, rod diameter and pitch), but a 1/2-scale height. In other portions of the facility, both diameter and height are subscale.

The scaled-height effects on void distribution are illustrated by analyzing transient level swell in a simple-geometry system. The model consists of a 0.1-m diameter vertical vessel partially filled with saturated water. A pressure transient, similar to the ROSA-III vessel depressurization transient following the opening of ADS, is imposed, and the responding mixture level swell is calculated. A drift flux model\(^{46}\), which has been tested against data from similar test geometry and transients, was used for this calculation. Figures 21.6 and 21.7 compare analytical results for full-height (10 m) and 1/2-height (5 m) cylinders of the same diameter. Lower-than-scaled swell level and lower void fraction are obtained in the 1/2-height cylinder than in the full-height cylinder. Lower void fractions are obtained in a height-scaled system because vapor volumetric flux, at a certain scaled height, is height-scaled when the same pressure transient is imposed. The deviation from the height-scaled swell level is larger for a faster depressurization. However, as shown in the present example calculation, the scaling error in the swell level in a 1/2-height system is acceptably small even during such a fast depressurization transient as immediately after the initiation of ADS. Also, it should be noted the scaled-height effects in ROSA-III are more or less compensated by the excessive vapor generation below the mixture level due to greater-than-scaled stored heat release from structures.

Flow-path characteristic diameter also influences mixture level swell and void fraction. Hence, the level swell in the ROSA-III vessel is affected by the lower-plenum and upper-plenum diameters and the downcomer annulus gap which are smaller than those in a BWR, although larger than any other full-pressure BWR simulation facilities. Generally, interphase slip velocity increases with diameter, and thus level swell becomes less. Thus, the effects of scaled diameter and scaled height tend to compensate each other. However, existing experimental data show the diameter effects are small for diameters larger than several centimeters, and no existing correlations or codes are reliable in predicting the diameter effects on level swell for such large diameters as those of BWR upper and lower plena.

The lower plenum flashing (LPF) causes fluid expansion and results in flows directing to the core and jet pump discharge line. The partition of flows between these two paths depends on gravitational heads and flow losses in the core and jet pump regions. Since the ROSA-III jet pumps have larger reverse flow loss coefficients than that of the reference BWR, more flow is partitioned into the core than in the reference BWR. This tends to slow the loss of lower plenum mass inventory through the jet pumps, to delay the discovery of the jet pump discharge line, and thus to delay the CCFL breakdown at the core inlet.

The geometry of the connection between the lower plenum and the jet pump discharge line also affects the timing of the jet pump exit recovery. Since this connection in ROSA-III is located lower than the scaled elevation, the jet pump exit tends not to be uncovered, and the vapor generated in the lower plenum tends to flow preferentially into the core.

The LPF causes significant mass redistribution in the vessel. CCFL is observed at the core inlet and outlet after the initiation of LPF except in the experiments with very small breaks, and has important influence on core liquid inventory during the refill-reflood phase. The onset of CCFL is detected by formation of a vapor pocket below the flow area restriction. The core inlet CCFL occurs at the bundle side-entry orifice (SEO), and limits the core liquid drainage. The core outlet CCFL occurs at the upper tieplate, and limits the liquid downflow from the upper plenum into the uncovered core.
Flooding occurs when vapor updraft velocity exceeds a certain limit dependent on the flow geometry. Since the core-inlet vapor velocity depends on flashing and boiling rates in the lower plenum, the core inlet flooding occurs only when the vessel depressurizes faster than a certain rate. Thus, the core inlet flooding occurs in ROSA-III either during the vapor discharge phase of an intermediate- or large-break test, or when vapor is discharged concurrently through break and ADS, however, does not occur for very-small breaks (< 2%) where the break remains covered and vapor discharge takes place through ADS alone.

The ROSA-III core inlet and outlet areas (at side-entry orifice, LTP, and UTP) are sized at about 50% of the BWR areas, so that the same pressure drops as in BWR are obtained for a volumetrically-scaled core flow rate, i.e., 1/2-scale core velocity. With 50% flow areas, the transient volumetric fluxes at the orifices are preserved full-scale, if the vapor generation rate per fluid volume is preserved.

Such scaling of the core inlet and outlet flow area restrictions provides CCFL characteristics close to that of the reference BWR. Figure 21.8 shows ROSA-III upper tieplate CCFL data\textsuperscript{19} taken at 1.1 MPa system pressure. The ROSA-III data, although show a large scatter and are limited in number, agree with correlations based on atmospheric pressure steam-water test data for the prototypical upper tieplate geometry. Two correlations are compared with the data. The first is one proposed by Naitoh\textsuperscript{20} on the basis of single-bundle data and confirmed by Nagasaka\textsuperscript{21} to be applicable to multibundle geometry of the Eighteen Degree Sector Test Apparatus (ESTA)

\begin{equation}
 j_d^{1/4} D^{1/4} + j_l^{1/4} D^{1/4} = 0.46
\end{equation}

where \( j_d \) and \( j_l \) are dimensionless phasic volumetric fluxes defined as

\begin{equation}
 j_d^* = j_d (\rho_g / g D \Delta \rho)^{1/2}
\end{equation}

\begin{equation}
 j_l^* = j_l (\rho_g / g D \Delta \rho)^{1/2}
\end{equation}

where \( j \) is the phasic volumetric flux, \( \rho \) the density, and \( g \) the acceleration due to gravity. The second correlation is one proposed by Sun\textsuperscript{22} on the basis of data of Jones\textsuperscript{23} and Naitoh\textsuperscript{19}

\begin{equation}
 K_g^{1/2} + K_l^{1/2} = 2.08
\end{equation}

where \( K_g \) and \( K_l \) are the dimensionless phasic volumetric fluxes defined as

\begin{equation}
 K_g = j_d (\rho_g / g D \Delta \rho)^{1/4}
\end{equation}

\begin{equation}
 K_l = j_l (\rho_g / g D \Delta \rho)^{1/4}
\end{equation}

and \( \sigma \) the surface tension.

The good agreement between the ROSA-III data and these correlations indicate that the upper tieplate CCFL characteristics is given in terms of vapor and liquid superficial velocities through the holes, which are approximately preserved full-scale in ROSA-III, irrespective of the hole diameter.

The CCFL characteristics of the ROSA-III side-entry orifice has not been measured. However, existing data taken for different diameters show only small influence of the orifice diameter on CCFL characteristics. Data taken in the Refill/Reflood Test Facility (RTF)\textsuperscript{24} show negligibly small influence of orifice diameter, \( D \), on \( j_d^* \) versus \( j_l^* \) plot, yielding a correlation:

\begin{equation}
 j_d^{1/4} / D^{1/4} + 0.8 j_l^{1/4} / D^{1/4} = 0.366 (g \Delta \rho)^{1/4}
\end{equation}

whereas data taken in the Two Loop Test Apparatus (TLTA)\textsuperscript{22} show a small but nonnegligible influence of \( D \), which can be represented by:

\begin{equation}
 K_g^{1/2} + 0.59 K_l^{1/2} = 2.14 - 0.0080 \pi D (\sigma / g \Delta \rho)^{-1/2}
\end{equation}
Fig. 21.9 depicts comparison of the side-entry orifice CCFL characteristics of ROSA-III (0.044 m diameter) and the reference BWR (0.0616 m diameter) predicted by these correlations. The two correlations agree well for these orifice diameters, and show that the ROSA-III side-entry orifice has almost the same CCFL characteristics as that of the reference BWR.

Although the counter-current flow interaction characteristics at the core inlet and outlet is approximately simulated by ROSA-III as discussed in the above paragraphs, the interphase momentum exchange inside the uncovered core may not be simulated well, since vapor volumetric flux therein is 1/2 scale. This should be considered in interpreting the level swell and heat transfer data inside the core.

21.7 Similarity between ROSA-III and BWR Rod Temperature Responses

The primary objective of an light water reactor (LWR) safety analysis concerning a LOCA is to confirm that the fuel rod surface temperature does not exceed the licensing acceptance criteria (1473 K). Thus, the core rod temperature response is of primary interest also in loss-of-coolant experiments. During a BWR LOCA, departure from nucleate boiling (DNB) (or “boiling transition” in BWR-specific technical term) on a fuel rod may occur either due to a power-cooling mismatch (i.e., flow stagnation) in a submerged core, or as a result of core uncover into steam environment. However, the former situation will occur only in limited occasions during the early phase of a large-break LOCA, and will be terminated soon by the core flow increase following LPF, before causing significant rod surface heatup. Therefore, the core uncover situations are of more importance. For these situations the core power generation rate is much less than the nominal operating power, and thus the radial temperature distribution inside the rod is almost uniform. This is true for both nuclear fuel rods and electric heater rods, despite the different rod-material thermal properties and different radial heat generation distributions. Then, the heatup rate of an uncovered rod is determined approximately, for both the nuclear fuel rods and heater rods, by the rod volumetric heat capacity, core power, and post-dryout core heat transfer rates. The energy balance of a heater rod (per unit length) with a rod surface superheating $T$ above the fluid saturation temperature $T_{sat}$ is

$$\Delta T \rho V C_p = \dot{Q}_{in} - \Delta T A h$$

(9)

where $\rho V C_p$ is the heat capacity, $\dot{Q}_{in}$ the core power, $A h$ is heat transfer rate due to convection and radiation, per unit length, respectively. As a rough approximation, one can assume that $\dot{Q}_{in}, h$ and the fluid saturation temperature $T_{sat}$ are constant, or changes only slowly with time. Then, Eq. (9) can be integrated easily to yield, for an initial condition of $\Delta T=0$ at $t=t_i$,

$$\Delta T = \left( \dot{Q}_{in} / A h \right) \ln \left( 1 - \exp \left[ \left( A h / \rho V C_p \right) (t - t_i) \right] \right)$$

(10)

Namely, the rod temperature approaches asymptotically to a final steady-state temperature $T = (\dot{Q}_{in} / A h) - T_{sat}$, which may exceed the licensing criteria of 1473 K, unless heatup is terminated by ECCS. Thus, to simulate the rod temperature rise in the reference BWR, it is necessary to simulate the rod heat capacity, heat transfer rate, as well as the rod dryout and rewetting timings. The primary material of the ROSA-III heater rod is boron nitride (BN) which is used as an electrical insulator between the cladding and heating element (see Fig. 17.2), and has similar volumetric heat capacity as that of uranium dioxide (UO$_2$) pellets in the nuclear fuel rod. For the rod heatup temperature range (between 600 and 1000 K) the heat capacity of the ROSA-III rod is between 120 and 140% of that of the BWR nuclear fuel rod (Fig. 21.10).
The post-dryout rod temperature can be estimated roughly by assuming that there is no heat transfer from the rod to the vapor environment until the rod is finally submerged by two-phase mixture. This approach was taken in licensing analyses until recently, since it gives a conservative estimation of PCT.

Realistically, the post-dryout heat transfer has a considerable influence on the rod surface temperature. The radiation heat transfer as well as the natural and forced convection heat transfer to steam, with the effects of liquid droplets, contribute the post-dryout cooling. The liquid effect is important at core upper elevations, where liquid downflow from the upper plenum improves core heat transfer. Therefore, the rod temperature shows "two-step" heatup behavior in experiments in which the failure of HPCS is assumed. The core heatup becomes much slower when LPCS is initiated as shown in Fig. 21.11, which compares rod temperature responses for different sizes of break in the recirculation pump suction line. For smaller breaks, the increase of core heat transfer due to LPCS leads immediately to rod temperature turnaround, because of relatively small core power at the time of LPCS initiation.

Empirical post-dryout heat transfer coefficients have been obtained from the ROSA-III rod surface temperature measurements above the mixture level (see Section 17). Heat transfer coefficients were obtained for two periods: before and after the actuation of LPCS. (The correlations were developed from experiments conducted without HPCS.) The pre-LPCS experimental data show that heat transfer coefficient (evaluated by dividing the convective heat flux by the rod surface superheat above the saturation temperature, $\Delta T = T - T_{sat}$) increases with pressure (see Fig. 17.6), and this is explained to be the result of the increase in the core vapor flow with pressure, caused by the increase in break flow with pressure. Thus, to extrapolate this pre-LPCS heat transfer coefficient to an actual BWR, the difference between the ROSA-III and BWR core vapor velocities should be considered. The core vapor velocity is smaller in ROSA-III than in the reference BWR because of two reasons: (i) the 1/212-scale core flow area (vs. 1/424-scale break area), and, for small breaks, (ii) the 60% ADS capacity. Thus, other conditions assumed to be the same, the core velocity in ROSA-III is 50% of that in the reference BWR for large breaks, and 30% for small breaks. It should be noted, in such scaling consideration, that the ROSA-III empirical heat transfer coefficients are defined for the wall superheating relative to the saturation temperature, rather than the actual (superheated) vapor temperature. The vapor velocity affects both (actual) heat transfer coefficient and vapor temperature.

Also the post-LPCS heat transfer coefficient may be affected by the smaller core vapor velocity. The vapor velocity is important to both the core exit CCFL and the in-core momentum exchange between vapor and liquid. For small breaks, the initiation of LPCS causes a pronounced increase in core heat transfer coefficient. This is reasonable since the vapor superficial velocities through UTP estimated for these experiments are less than the flooding limit which can be obtained by putting $\dot{j}_i = 0$ in Eqs. (1) and (4); LPCS water falls into the core. At the LPCS initiation pressure of 2.2 MPa, the estimated vapor superficial velocity through the ROSA-III UTP is about 3 m/s (when the core vapor flow rate is assumed to be equal to the total vapor flow rate through the break and ADS) versus the flooding limit obtained from Eqs. (1) or (4) of about 6 m/s. The predicted vapor superficial velocity through the BWR UTP is 5 m/s; the ROSA-III vapor velocity is smaller because of the 60% ADS capacity. Thus, it would be reasonable to consider that the ROSA-III post LPCS heat transfer coefficients exaggerate the spray effects on core heat transfer for small breaks. For large breaks, the vapor-convection heat transfer coefficient is still high at the time of LPCS initiation, and thus the effect of LPCS on core heat transfer is not clearly seen in the experimental data.
21.8 Conclusions

The ROSA-III experiments were designed to simulate, on a real-time basis, full-scale pressure and temperature transients during a BWR LOCA, by using a volumetrically- and height-scaled facility. Good simulation capability of the ROSA-III facility has been concluded for a variety of LOCA transients, on the basis of computer code prediction of BWR thermal-hydraulic responses. The sequence and timings of major events, as well as the transient responses of pressure, mixture level, and core rod-surface temperatures, are simulated well by ROSA-III. The interrelationship among the responses of these parameters as well as the scaling and scaling-distortion effects on the responses of ROSA-III have been studied by analyzing the ROSA-III experimental data. The conclusions derived from these similarity studies are:

(1) ROSA-III closely simulates the vessel pressure response of the reference BWR. The difference between the responses of the two systems occurs primarily because of the slower downcomer level drop in ROSA-III than in the reference BWR for very small (e.g., <1%) recirculation-line breaks, and because of excessive stored heat release in ROSA-III for large breaks. For instance, the core mixture level swell following the lower plenum flashing is less significant in ROSA-III than in the reference BWR.

(2) The ROSA-III vessel mixture level behavior agrees well with that in the reference BWR. However, the ROSA-III core mixture level is affected by the scaled heights, scaled diameters, and excessive stored heat release from structures. For large breaks, the atypical dynamic pressure losses in the jet pump and atypical lower plenum geometry also affect the core mixture level. The core-inlet and outlet CCFL conditions seem to be simulated reasonably well.

(3) ROSA-III represents the rod temperature response of the reference BWR. The core mixture level, which dictates the rod temperature response, is well simulated. Difference is found between the ROSA-III and reference BWR rod temperature responses for large breaks, as the core mixture level response is not so well simulated as for small breaks. For instance, the core mixture level swell following the lower plenum flashing is less significant in ROSA-III than in the reference BWR.

(4) The approach taken in this report have proved useful for examining the similarity between the ROSA-III and BWR thermal-hydraulic responses. However, its usefulness is limited by the computer codes’ ability to predict the BWR responses during a LOCA.

References


14) Suzuki, M., et al.: Similarity study of large steam line break LOCAs in ROSA-III, FIST and BWR/6, to be published.

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**Fig. 21.11** Break area dependence of rod temperature response to recirculation line break (shown for locations where PCT was observed in individual tests).
22. Conclusions

The ROSA-III Program has conducted more than one hundred runs of loss-of-coolant experiments simulating a broad spectrum of accidental conditions. These tests involved series of sensitivity tests which investigated the effects of important test parameters, including break size, break location, emergency core cooling system (ECCS) failure mode and ECCS trip logics, core power decay characteristics and spatial power distribution, as well as emergency core coolant (ECC) temperature.

The experimental results have been interpreted and analyzed in detail. The analyses included comparisons between the computer code-predicted thermal-hydraulic responses of ROSA-III and the reference BWR (Chapters 3, 6, 7, 9, 12, 18, 19, 20 and 21). These comparisons confirmed that ROSA-III can simulate well the overall responses of the reference BWR. Also, the counterpart tests performed at the ROSA-III and FIST facilities indicated qualitatively similar responses of the two facilities although the two facilities took different scaling approaches to simulate the BWR geometry (Chapters 18 through 20).

The most important conclusions derived from these experimental and analytical studies are:

(1) The accidental scenario of a BWR LOCA has been well understood through the experiments. The sequence and timings of major events during a LOCA have been defined experimentally for various break locations and break sizes.

(1-a) The overall system responses are not much affected by the size of break when break occurs in the recirculation line (Chapter 5). The system is depressurized as vapor is discharged from the vessel. The vapor discharge begins either as the downcomer empties to uncover the broken recirculation line nozzle (for large breaks), or as the downcomer level drop trips on the automatic depressurization system (ADS) (for small breaks). With the failure of the high pressure core spray (HPCS), the core once uncovers to steam environment, but becomes submerged again as the system depressurizes to begin the injection of the low-pressure ECCSs. pressure ECCSs.

(1-b) The discharge flow rate out of a break located in the recirculation line is limited by the two flow area restrictions, which are located at the jet pump drive nozzle and the recirculation pump discharge in the broken recirculation line respectively. The total area $S$ of these area restrictions is about 80% of the maximum recirculation line flow area. Then, if the break area is smaller than $S$, the system response is irrespective of whether the break is located at the suction side or discharge side of the pump. Also, the maximum effective break area is limited to $S$ when the break is located at the pump discharge side (Chapters 14 and 16).

(1-c) A steam line break causes a rapid system depressurization. Thus a significant level swell occurs in the downcomer, and low-downcomer-level trip signal for ECCS is generated late in the transient. Then, if both HPCS and the containment overpressure trip for ECCS should fail in a BWR steam-line break LOCA, the initiation of ECCS delays considerably, and this ECCS delay may possibly cause core uncovery and heatup (Chapter 15).

(2) The effectiveness of the current BWR ECCS design has been well demonstrated, and the impact of postulated ECCS failure on core cooling has been evaluated.
(2-a) Among the possible ECCS single failure modes, the failure of HPCS has the most serious influence on core cooling, for large as well as small breaks located in the recirculation line (Chapter 3 and Appendix 2).

(2-b) The PCTs obtained in the experiments were considerably lower than the current licensing criteria of 1473 K; for all the break sizes, break locations, and ECCS failure modes investigated (Chapters 3 through 21 and Appendix 2).

(2-c) The failure of pressure control system, an increase in ECC temperature, or earlier ADS actuation alters little the performance of ECCS (Chapters 7, 8 and 13).

(3) Close correspondence has been observed between the core mixture level and heater rod surface temperature responses.

(3-a) Major temperature excursion of the core heater rods occurs only when the rod surface uncoovers above the mixture level. The rod surface is quenched when it becomes submerged again by two-phase mixture.

(3-b) The PCT is determined primarily by the duration of dryout (i.e., the duration of core uncovercy) and the core power. However, the post-dryout cooling, by steam flow and liquid droplets in the core, has considerable influence on the temperature response of uncovered rods (Chapter 17).

(4) Several new findings have been obtained concerning the BWR thermal-hydraulic responses during a LOCA.

(4-a) The most serious core cooling conditions are not necessarily encountered for the maximum break size; smaller break sizes may result in higher PCTs in certain circumstances. For the recirculation pump suction-line break tests with an assumed failure of HPCS, the maximum PCT was obtained for a scaled break area of 50% (Chapter 5).

(4-b) A large break (greater than 50%) on the recirculation pump discharge side may result in a higher PCT than the same size of break on the pump suction side.

(4-c) Fluid remaining in the feedwater line begins to flash when the vessel depressurizes to the fluid saturation pressure (2.2 MPa). This flashing holds up the vessel pressure and delays the initiation of LPC1, which has a starting setpoint pressure of 1.65 MPa (Chapters 3, 4 and 5).

(5) The natural circulation core cooling mode was simulated in separate effects tests. These tests showed sufficient core cooling ability for decay power when the core is submerged below the mixture level. The internal natural circulation mode through the core bypass has been successfully modeled to predict the in-shroud mixture level for given pressure, core power and downcomer level (Chapter 9).

(6) The ROSA-III experiments have provided a significant data base for development, improvement and assessment of computer codes for BWR LOCA analyses. The data have already been used widely both inside and outside JAERI for assessment of codes including: a BWR licensing evaluation code SAFER, advanced codes with non-equilibrium non-homogeneous two-phase flow models like TRAC-BD1 and RELAP5/Mod1, and conventional or simplified-model LOCA analysis codes like RELAP4/Mod6 and THYDE-B1. The particularly important code assessment effort that has been made with ROSA-III data has been the OECD/NEA CSN1's Twelfth International Standard Problem (ISP-12) which used the data from a ROSA-III 5%-break experiment Run 912 (Chapter 6). This ISP was the first ISP on a BWR LOCA experiment, and thus provided an important opportunity to the international participants who were interested in assessing computer codes' ability in predicting the BWR thermal-hydraulic response during a LOCA.
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The authors are indebted to Messrs. H. Asahi, T. Odaira, T. Takayasu, S. Sekiguchi, Y. Kitano and T. Numata of Nuclear Engineering Corporation for their assistance in performing the experiments and Miss T. Kurosawa and Miss M. Kikuchi of Nihon Computer Bureau Corporation for typing the manuscript.
Appendix I  Characteristics of ROSA-III Facility

1.1 Major Changes of ROSA-III Facility

The ROSA-III facility was modified to improve the characteristics and extend the capability. Major changes of the facility are listed in Table A1.1. These information are important to understand the conditions of each test listed in Appendix II. Four simulated fuel assemblies were used in the ROSA-III experiments. Table A1.2 summarizes the major specifications of the simulated fuel assemblies.

1.2 Pump Characteristics

The single-phase homologous head and torque curves for the ROSA-III recirculation pump in shown in Figs. A1.1 and A1.2, respectively. The ROSA-III jet pump characteristic is shown in Fig. A1.3. The HPCS, LPCI and LPCI pump characteristics are shown in Figs. A1.4, A1.5 and A1.6, respectively.

1.3 Heat Loss Characteristics

The net heat loss, \( Q_{\text{HL}} \), of the ROSA-III facility for the temperature difference, \( \Delta T \), between system and atmospheric temperatures is characterized by the following simple expression, as shown in Fig. A1.7.

\[
Q_{\text{HL}} = 0.56 \Delta T
\]

1.4 Pressure Loss Characteristics

The local pressure losses were measured for the system with installation of the simulated fuel assembly No. 4. Table A1.3 shows the pressure loss coefficients which are defined by the following expression.

\[
K = \frac{\Delta P}{2 \rho_f g A_f^{\frac{3}{2}}}
\]

where \( \Delta P \) is local pressure loss which includes frictional and form losses across the measuring span, \( W_f \) is local mass flow rate, \( A_f \) is minimum flow area through the measuring span, \( \rho_f \) is fluid density, \( g \) is acceleration of gravity.

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\[ Q_r = 450 \, \text{l/min}, \quad V = Q / Q_r \]
\[ \omega_r = 3600 \, \text{rpm}, \quad \alpha = \omega / \omega_r \]
\[ h_r = 262 \, \text{m}, \quad h = H / H_r \]

Fig. A1.2 Single-phase homologous torque curve for the ROSA-III recirculation pump
\[ Q_r = 450 \, \text{l/min}, \quad V = Q / Q_r \]
\[ \omega_r = 3600 \, \text{rpm}, \quad \alpha = \omega / \omega_r \]
\[ T_r = 184 \, \text{Nm}, \quad \beta = T / T_r \]

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### Table A1.2  ROSA-III simulated fuel assemblies

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### Table A1.3  Pressure loss coefficients (Re > 10⁵)

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<td>Channel inlet orifice (hole of 44.0 dia. per channel)</td>
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Fig. A1.1 Single-phase homologous head curve for the ROSA-III recirculation pump.

\[ Q_s = 450 \, \ell/\text{min}, \quad V = Q/Q_s, \]
\[ \omega_s = 3600 \, \text{rpm}, \quad \alpha = \omega/\omega_s, \]
\[ H_s = 262 \, \text{m}, \quad h = H/H_s, \]
Fig. A1.2 Single-phase homologous torque curve for the ROSA-III recirculation pump.

\[ Q_r = 450 \text{ l/min}, \quad V = Q/Q_r \]
\[ \omega_r = 3600 \text{ rpm}, \quad \alpha = \omega/\omega_r \]
\[ T_r = 184 \text{ Nm}, \quad \beta = T/T_r \]
Fig. A1.3  ROSA-III jet pump characteristic.

Fig. A1.4  HPCS pump characteristic.
Fig. A1.5  LPCS pump characteristic.

Fig. A1.6  LPCI pump characteristic.
Fig. A1.7  ROSA-III heat loss characteristic.
### Appendix II  Summary Table of ROSA-III Tests

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<th>Core Inlet Subcooling ∆T (K)</th>
<th>U Plenum Shape (%)</th>
<th>Fuel Assembly No.</th>
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**Remarks:**
- Fuel acceptance test
- QSV was kept open
- Split break
- Isothermal blowdown test
- Power trip at 156 s
- Power trip at 220 s
- Steam line valve leakage
- ADS, ECC timing error
- HPCS timing error
- Power trip at 372 s
- Partial power trip at 114 s, ADS timing error
- Early ADS trip
- Early ADS trip
- Delayed MSIV closure
- No.4 fuel acceptance test, Small MSIV area
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(1)  Location: RS - Rockville; L1 - Lightwater 1; L2 - Lightwater 2
(2)  Area: 200 - 200 kW; 50 - 50 kW
(3)  U. Pienum Quality: A-2, D, B-1
(4)  Power Curve: A-2, D, B-1
(5)  Trip Time: 6.4 126 s, 11.5 129 s, 8.3  All s
(6)  ECCS Actuation Time: 13.6 134 s, 6.5 129 s, 8.7 129 s

Remarks: Pump trip at 101 s, MSIV trip on L1+3, ADS delay, PCS became operable as following test.
### Summary Table of ROSA-III Tests (Cont’d)

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<th>Power Curve</th>
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**Note:**
1. Break location:
   - RS: Recirculation pump suction line
   - RD: Recirculation pump discharge line
   - SL: Main steam line
   - SLO: Main steam line outside the containment vessel

2. Percent of 1/424 scaled BWR pipe flow area. Flow area of main steam line is 1.4 times of the recirculation line flow area in the BWR/6 (251-848) system.

3. Average quality in upper plenum is calculated from the total core power, total core flow rate and core inlet enthalpy.

4. Nine different power curves were used depending on the test objectives.
   - A-3: Delayed fusion power + FP decay power
   - B-1: Power curve with transient void feedback effect
   - D: APS decay curve

5. Standard trip logic for MSIV and ADS:
   - MSIV: L-2+3 se
   - ADS: L-1+20 s

6. Standard ECCS actuation logic:
   - HPSC: L-1+40 s and P ≤ 2.2 MPa
   - LPC: L-1+40 s and P ≤ 1.6 MPa

7. Failed ECCS:
   - H: HPSC, S: LPCS, I: LPCI
Appendix III  Base Test Series for a Large Break Loss-of-Coolant Accident

III.1 Introduction

Six large break LOCA tests\textsuperscript{13) - 13)} were conducted with varying break conditions including a double-ended break in the largest diameter piping of the primary cooling system. The test conditions are:

- **Run 701.** double-ended break with the ECCS actuation and the decay heat simulation
- **Run 702.** 200% split break without ECCS actuation
- **Run 703.** 100% split break with ECCS actuation
- **Run 704.** double-ended break with ECCS actuation
- **Run 705.** double-ended break with no heat generation in the core and without ECCS actuation
- **Run 706.** double-ended break without ECCS actuation.

Run 704 is a baseline test for large diameter break LOCAs with a break located at the pump inlet of the recirculation line. The initial fluid conditions and other operating conditions such as the main steam line closure, feedwater line closure, and ECCS actuation also simulate those of the BWR/6. The test conditions of the other five tests are based on the test condition of Run 704 with modifications in the break mode, break size, core decay power, and ECCS actuation conditions. Through comparisons of the test results of these tests, it is possible to evaluate the separate effect of each of these test parameters to obtain a better understanding of the LOCA phenomena.

An analysis was made using the RELAP5/MOD0 code to predict the thermohydraulic transient for the baseline test and evaluate the predictive ability of the code based on a two-fluid model.

III.2 Experimental Procedure and Conditions

III.2.1 Procedure

After the initial fluid conditions were established, the rupture disks were broken and the core power was reduced in accordance with a predetermined decay curve. The volume of each component of the ROSA-III facility is scaled to 1/424 of the BWR/6 system. Accordingly, the core power before initiation of the break should be 9 MW to produce the same transient expected in the BWR/6 system during a LOCA. The available core power is 4.2 MW. Compromise was made to keep the core power at 4.2 MW for 10 s after the break until the calculated core power reduced to 4.2 MW. The core power was reduced thereafter as the stored heat and decay heat decreased with time.

The ECCS was actuated automatically and injected water into the pressure vessel as soon as the actuation condition such as the low level in the downcomer was reached.

III.2.2 Conditions

All the tests in this test series were conducted with the same initial conditions, for example, the system pressure of 7.1 MPa in the steam dome, and the core flow rate of 36 kg/s. The power supply to the core was 3.7 MW in Runs 701, 702, and 703, and 3.3 MW in...
Runs 704, 705, and 706. The heat generation rate of each fuel rod in the core was the same with chopped cosine distribution along the axis with a peaking factor of 1.41. The maximum linear heat generation rate was 11 kW/m in Runs 701, 702, and 703, and 10 kW/m in Runs 704, 705, and 706.

The main steam line and the feedwater line were closed following the break with a time delay predetermined by the analysis to best simulate the main steam line and the feedwater line closures during an accident in a BWR/6. The HPCS was actuated to provide the cooling water at 27 s after the break and the LPCS was actuated when the pressure decreased to 2.1 MPa in the pressure vessel. The LPCI was started 13 s after the LPCS was started in all the tests.

Run 704 is a baseline test with the double-ended break condition at the pump inlet side of the recirculation line. The initial fluid condition, the main steam line, and the feedwater line closures were simulated. All the ECCSs were actuated.

Run 701 is a double-ended break test with ECCS actuation and with the reduced core power generating the decay heat only. The stored heat was neglected. Accordingly, the effect of the stored heat on the thermohydraulic response of the primary cooling system and the core could be evaluated through comparison of the results with those of Run 704.

Runs 702 and 703 are the split break tests with the break size of 200 and 100%, respectively. The 100% break means the break area equal to the cross-sectional area normal to the axis of the largest piping of the primary cooling system. The effect of break mode and break size could be evaluated through comparison of the test results of Runs 702, 703, and 704.

Run 705 is a simple isothermal blowdown test without heat generation in the core. The test data give fundamental blowdown characteristics of the facility and make it possible to understand the hydraulic behavior during a LOCA and also to assess the computer code.

Run 706 differs from Run 704 as the main steam line and the feedwater line were closed simultaneously at the initiation of the break. The ECCS was not actuated in this test. Accordingly, the effect of the ECCS and the main steam line and feedwater line closure could be evaluated through comparison of the test results with those of Run 704. The primary test conditions are summarized in Table 1.

III.3 Experimental Results

The system pressure in the baseline test Run 704 (see Fig. 1) decreases after break due to the discharge of fluid through the break. The pressure starts to increase after closure of the main steam isolation valve (MSIV) at 4.5 s after break. The mixture level in the downcomer decreases rapidly after break and reaches the outlet nozzle to the recirculation loop at 12 s after break. Then the steam in the vessel discharges directly through the vessel side break and the system pressure starts to decrease more rapidly. The lower plenum fluid saturates at 17 s after break due to the decrease in the system pressure, and flashing is initiated. However, the rate of the system pressure decrease slows down after the initiation of the lower plenum flashing (LPF).

The fuel rod surface temperature in Run 704 is shown in Fig. 2 and is compared to the results of Run 706 without ECCS actuation. The measured fuel surface temperature starts to increase rapidly at 13 s after break due to boiling transition for the top 70% of the core. The fuel surface temperature exhibits rewetting after 17 s by the LPF below the midplane of the core. In Run 704, the spray of water from the HPCS and LPCS systems limits the rate of increase in the fuel surface temperature at the upper portion of the core.
After initiation of the high flow rate LPCI system at 80 s, the mixture level in the vessel recovers fast and the fuel surface temperature turns around and decreases. The whole core is reflooded at 95 s. The fuel rods are quenched from the bottom to the top and the whole core is quenched at 120 s after break.

The mixture level transients in the core and lower plenum measured by conductivity probes are shown in Fig. 3 for (a) the 200% double-ended break tests of Run 704 with ECCS actuation and Run 706 without ECCS actuation and (b) the 100% split break test of Run 703 with ECCS actuation. The trend of mixture level transient in Run 703 is similar to that in Run 704. Mixture level in the core falls quite fast from 13 to 15 s after break and recovers due to the LPF between 17 and 30 s. After 30 s, a separate mixture level is formed in the lower plenum indicating occurrence of CCFL at the core inlet orifice. Both mixture levels in the core and lower plenum fall with time. The lowering rate in the core in Run 704 is slower than that in Run 706 without ECCS actuation, indicating the accumulation of water from the HPCS in the core. In the lower plenum, the accumulation is negligible. After initiation of the LPCI at 80 s after break, the mixture levels in the core and lower plenum recover. The recovery rate in the core is especially large due to level swell by increased steam generation in the core after actuation of the LPCI. The fluid behavior can be estimated by the comparison of fluid temperatures below and above the tie plate. Fluid temperatures above the upper tie plate measured at ten locations exhibit a superheated condition after break and returns to saturation temperature a few seconds after actuation of the HPCS, showing mixing of subcooled HPCS water with steam from the core and accumulation of two-phase fluid in the upper plenum. The fluid temperature below the upper tie plate exhibits a superheated condition from 17 s after break and returns to saturation temperature at 88 s after break, 8 s after LPCI initiation at 80 s, indicating levels well after 88 s. Between 40 and 80 s, fluid temperatures at several locations below the upper tie plate exhibit a saturation temperature, showing that some portion of the liquid accumulated in the upper plenum falls down into the core.

There is a strong correlation between the mixture level transient and the fuel rod surface temperature behavior in the core, as shown in Fig. 3 for Run 704. In the figure, the mixture level transient measured by conductivity probes is compared with the dryout and rewetting behavior of heater rods detected by thermocouples embedded in the cladding of simulated fuel rods. Between 13 and 15 s after break, the fuel surface temperature excursion occurs from the top following the decrease in the mixture level height. Between 17 and 20 s, the fuel surface rewets at positions 5 and 4 (core midplane) following the increase in the mixture level height by the LPF. The midplane of the core dries out again from 35 to 84 s following the mitigation of LPF and the decrease in the mixture level height. The final quenching of the core occurs by LPCI from bottom to top, delayed from the reflooding of the core by as much as 25 s.

The system pressure and fuel rod surface temperatures measured in Runs 701 through 706 are shown in Fig. 4. The trends of system pressure transients in Runs 702 and 703 are similar to those of the standard test, Run 704: initial decrease after the break, recovery by closure of the MSIV, rapid decrease by uncovering the outlet nozzle to the recirculation loop, and decrease in the depressurization rate by the initiation of LPF. The depressurization is fastest in Run 702 with a 200% split break, possibly due to the largest discharge flow among the three test Runs 702, 703, and 704. The depressurization rate of Run 704 with a 200% double-ended break is slower than that of Run 702 and faster than that of Run 703 with a 100% split break, indicating that the break flow by a 200% double-ended break at the recirculation pump inlet line is less than that by a 200% split break and greater than that
by a 100% split break.

In Run 706 without ECCS actuation, the steam line and feedwater line are closed at the time of break, and the system pressure is kept almost constant after break due to the balance of steam generation in the core and discharge flow through the break. The system pressure decreases rapidly after uncovering of the recirculation line outlet nozzle, becoming similar to the depressurization behavior of the baseline test, Run 704.

In the decay heat simulation test, Run 701, the system pressure decreases monotonously after break without recovery by the MSIV closure because the heat generation in the core is small after break due to the decay heat simulation and the effect of break flow to the system pressure is greater than the effect of steam generation in the core even after MSIV closure.

In the isothermal blowdown test, Run 705, the system pressure decreases slowly after break, although the steam line is closed at the time of break, because there is no power supply to heater rods in the core in Run 705. The depressurization is accelerated by uncovering of the recirculation line outlet in the downcomer; however, slowdown of the depressurization rate by LPF is not observed in the experimental data because the whole system is nearly saturated from the beginning of blowdown in the isothermal blowdown test, Run 705.

The trends of fuel surface temperature transients in Runs 702, 703, and 706 are similar to those in the baseline test, Run 704 before HPCS actuation at 27 s after break. The fuel rod surface temperature starts to increase by the boiling transition from the top of the core following the decrease in the mixture level height in the core. The fuel surface rewets by the mixture level rise due to LPF. The effect of LPF is the most prominent in Run 702 with a 200% split break and the whole core rewets by the LPF because the depressurization rate before the initiation of LPF is largest in Run 702. The core dries out again from the top following the fall of the mixture level due to mitigation of LPF. A large difference is seen in the fuel surface temperature transient after actuation of the ECCS. The whole core dries out in Run 702 and 706 without ECCS actuation and the fuel rod surface temperatures keep rising until the power supply to the core is terminated to protect the core. The lowest part of the core is not uncovered to steam environment in Runs 703 and 704 with ECCS actuation due to accumulation of part of the sprayed water from the HPCS and LPCS in the bottom part of the core. The whole core is reflooded and quenched from bottom to top in Runs 703 and 704 after initiation of the large flow rate LPCI at 92 and 80 s after break in Runs 703 and 704, respectively.

The peak cladding temperature (PCT) in the double-ended break test, Run 704, was 774 K; it occurred at position 3 (588 mm from the top of the active core) at 85 s after break. The PCT in a 100% split break test Run 703 was 761 K and occurred at position 2 (353 mm from the top of active core) at 97 s after break. Therefore, the PCT is a double-ended break test was 13 K higher than the PCT in a 100% split break test.

In the low power test, Run 701, the system pressure decreases fast and the LPF starts at 12 s after break, before initiation of core uncovery; therefore, no fuel surface temperature rise is seen in Run 701 before 40 s. After 40 s, the trend of fuel surface temperature is similar to that in the standard test Run 704. The core uncovers from top to midplane following the decrease in the mixture level height due to mitigation of LPF; however, the temperature rise of the fuel surface is small due to low power input in Run 701. The core is reflooded from the bottom after initiation of the LPCI at 72 s after break and the whole core is quenched at 105 s after break.

In the isothermal blowdown test, Run 705, the fuel surface temperature decreases after break following the decrease in the saturation temperature of the system. The whole core
dries out from the top between 40 and 90 s after break following the decrease in the mixture level height without ECCS actuation. The fuel surface temperature becomes constant after dryout because of no heat generation in the fuel rod in isothermal blowdown test Run 705.

III.4 Analysis

The test results of the baseline test, Run 704, with full ECCS actuation were analyzed with the RELAP5/MOD0 code\(^{14}\). The RELAP5 code is an advanced, one-dimensional, fast-running system analysis code developed at the Idaho National Engineering Laboratory for the analysis of light water reactor LOCA and non-LOCA transients.

The hydrodynamic model is a five-field equation, two-fluid model consisting of two phasic continuity equations, two phasic momentum equations, and an overall energy equation. Heat transfer correlations are the same as those in the RELAP4/MOD6 code\(^{15}\).

There are few adjustable input parameters of RELAP5. Jet pump model, mixture level, and CCFL at the upper tie plate and core inlet orifice are very important in a BWR LOCA/ECCS analysis; however, momentum mixing in a jet pump is not accounted for accurately in the present version of RELAP5. The choked discharge flow is calculated by the RELAP5 break flow model based on the method of characteristics\(^{16}\). The calculated system pressure is compared with the experimental data in Fig. 5. The overall trend of calculated system pressure is in good agreement with the experimental data; however, the timing of recirculation line uncovering and initiation of LPF are delayed compared with the data, possibly due to the underprediction of break flow. Temporary recovery of the calculated pressure after 40 s is attributed to the increased steam generation rate in the core due to water falling from the upper plenum after the complete dryout of the core. No pressure recovery was observed in the experiment because the lower part of the core was always covered with the two-phase fluid, and CCFL at the upper tie plate limited the water fall from upper plenum.

The fuel surface temperatures calculated at the midplane (position 4) of the core are compared with the experimental results in Fig. 6. Boiling transition due to mixture level fall in the core before LPF is calculated accurately in the analysis and the timing of the boiling transition and the rate of fuel surface temperature increase agree well with the experimental results.

The fuel surface temperature can be closely correlated with the mixture level transient or void fraction change in the core. The calculated void fraction at the midplane of core is shown in Fig. 7. The void fraction increases after break, stabilizes at about 0.7, and increases rapidly to 1.0 at 15 s after break. At this time, the rise in fuel temperature is correctly calculated as shown in Fig. 6.

The lower plenum flashing occurs at 18 s after break in the calculation. The core inlet flow increases due to LPF as it is clearly shown in Fig. 8. This results in rewetting of the fuel surface in both prediction and experiment. In the calculation, however, the mixture level rises to the top of core and the entire core is quenched unlike the experiment.

As LPF subsides, the mixture level falls again and the whole core is uncovered in the analysis whereas the bottom part of the core is covered by a two-phase mixture throughout the transient in the experiment. As a result, the calculated fuel surface temperature deviates away from the experimental results after the initiation of LPF in the upper half of the core and after the level fall at about 40 s in the lower half of the core.
III.5 Conclusions

A series of large break tests were conducted at the ROSA-III test facility. The following conclusions were obtained through the examination and comparison of the test results.

(1) The ROSA-III test facility can simulate major aspects of a BWR LOCA: fuel surface temperature rise due to lowering of the mixture level in the core, rewetting by LPF, and final quenching by ECCS.

(2) The balance between the steam generation in the system and the steam discharge from the system plays a key role for the pressure transient during a BWR LOCA. Therefore, heat transfer to coolant in the core, LPF, closure of the MSIV, break flow, and the uncovering of the recirculation outlet nozzle in the downcomer are important factors in determining the pressure transient during a BWR LOCA.

(3) The fuel surface temperature rises first from the top of the core after the mixture level falls in the core, then drops to saturation temperature due to rewetting as the mixture level rises following LPF, but rises again due to the mixture level fall following the mitigation of LPF, and finally quenches by reflooding of the core by LPCI.

(4) The ECCS effectively cools the core during a large break LOCA. The HPCS and the LPCS limit the surface temperature increase. The LPCI provides a sufficiently large amount of water to reflow the core and quench the fuel surface.

The following conclusions were obtained through the analysis of the baseline test, Run 704, with the advanced code RELAP5/MOD0:

(1) The RELAP5 code's capability for analyzing a BWR LOCA is confirmed.

(2) Overall agreement between the calculated and measured system parameters is good, but there are some noteworthy differences that are attributed to the limitation of the present version of RELAP5/MOD0 to accurately calculate 1) CCFL at the upper tie plate and core inlet orifice, 2) momentum mixing in jet pumps, and 3) motion of mixture level.

References


<table>
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<tr>
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*a dp = Decay power of fission products and actinides.
*b dp = Delayed neutron fission power.
*c dp = Stored heat release from fuel rods.
Fig. 1  Steam dome pressure transient in standard test run 704.

Fig. 2  Fuel surface temperatures for the rod A33 in standard test run 704 with ECCS actuation and in run 706 without ECCS actuation.
Fig. 3 Mixture level transients inside shroud for (a) 200% double-ended break test run 704 with ECCS actuation and run 706 without ECCS actuation, and (b) 100% split break test run 703 with ECCS actuation.
Fig. 4  Pressure and fuel surface temperature transients in runs 701 through 706.
Fig. 5  Comparison of the calculated system pressure by RELAP5/MOD0 with the measured data in standard test run 704.

Fig. 6  Comparison of the calculated fuel surface temperature by RELAP5/MOD0 with the measured data at the midplane of the core in standard test run 704.
Fig. 7  Calculated void fraction at the center of the core.

Fig. 8  Calculated core inlet flow.
Appendix IV  Small Break Loss-of-Coolant Accident Simulation Tests

IV.1  Introduction

The break size of 5% was selected for the ROSA-III small break tests because the highest PCT for break area between 1 and 100%, was calculated\(^1\) for the 5% break by using the design code SAFE\(^2\). The 5% break area in the ROSA-III facility was 5% of (1/424) of the cross section of the BWR recirculation pump inlet line.

The objectives of the 5% small break LOCA test series were:

1. To understand the basic phenomena during a postulated 5% small break LOCA,
2. To evaluate ECCS performance during a 5% small break LOCA,
3. To evaluate the capability of a reactor safety analysis code (RELAP4/MOD6\(^3\)) to predict 5% small break LOCA behavior.

IV.2  Test Conditions

The test conditions for the 5% small break test series are summarized in Table 1. The primary differences among the tests were the ECCS conditions. In the baseline test, Run 807, all ECC systems were actuated as specified, whereas a single failure of the ECCS, requested in the acceptance criteria\(^4\) of the ECCS for nuclear power reactors, was assumed in Runs 8051, 8061 and 808. In order to compare the severity of each single failure assumption of the ECCS in a 5% small break LOCA, failures of the HPCS, LPCD-DG (diesel generator), and LPCS-DG were assumed in Runs 8051, 8061 and 808, respectively. There are three LPCI systems in the BWR/6 and the failure of the LPC-DG results in the failure of two of the three LPCI systems. The failure of the LPCS-DG results in the failure of the LPCS and one LPCI system. Because the HPCS and ADS are designed primarily to play key roles in a small break LOCA, sensitivity tests for the actuation time of the HPCS and ADS were conducted in Runs 804 and 809, respectively.

The break was assumed to be a 5% split break at the recirculation pump suction line. The break was simulated by an orifice with a diameter of 5.9 mm. Blowdown was initiated by opening the quick opening blowdown valve located immediately downstream of the orifice. The valve opening time is less than 0.1 s. The initial test conditions were as follows. The steam dome pressure was 7.3 MPa with the corresponding saturation temperature of 562 K. The core inlet flow rate was 16.0 kg/s and the core outlet quality was approximately 11%. The power supplied to the simulated fuel bundles at steady state was 3,554 kW.

The experimental procedure was as follows. Closing of the feedwater line was initiated at 2 s and completed at 4 s after break. Closing of the steam line was initiated at 27 s and completed at 31 s. The time to close the feedwater line valve was based on the evaluation guide\(^1\) for BWR safety. The time to close the steam discharge line valve and to actuate HPCS and ADS were based on the results of calculations for a 5% split break in a BWR using a BWR safety analysis code\(^5\).

The HPCS was actuated at 50 s after the break in Runs 8061, 807 and 808. In Run 804, HPCS actuation was delayed until 236 s after the break. The ADS valve was opened at 230 s in Run 804, with delayed actuation of the HPCS, and Run 8051, without HPCS actuation. The valve was opened at 330 s in Runs 8061, 807 and 808, with HPCS actuation,
and at 170 s in Run 809, for an ADS actuation time sensitivity test without the HPCS.

The system pressures for actuating the LPCS and LPCI were 2.2 and 1.6 MPa, respectively, and were based on the pump characteristics used in the safety analysis of a BWR\textsuperscript{13}. The measured system pressures at LPCS and LPCI initiation were a little higher than specified in the test conditions.

The test conditions of a 200% double-ended break test, Run 733, are also shown in Table 1. The characteristics of a 5% small break LOCA can be made clearer through comparison of the test results with those of a double-ended break LOCA test.

IV.3 Experimental Results

IV.3.1 Sequence of Events and System Pressure

The sequences of events for each experiment are summarized in Table 2. The break occurred at time zero when core power control was initiated and pump power was terminated. Feedwater line closure and steam line closure were initiated at 2 and 27 s, respectively. Due to the main steam line closure and the resultant decrease of steam discharge flow, the system pressure recovered as shown in Fig. 1. The system pressure started to drop again after HPCS actuation at 50 s due to condensation of steam in Runs 8061, 807 and 808, and after initiation of dryout at 129 s and the resultant decrease in steam generation rate in the core in Runs 804, 8051 and 809 without HPCS actuation. The pressure decrease rate was accelerated by the actuation of ADS regardless of HPCS actuation.

The liquid level in the downcomer decreased to the elevation of jet pump suction and the recirculation pump suction, at 129 and 212 s, respectively, in Run 8051 (without HPCS actuation). The uncovering of the recirculation pump suction nozzle resulted in the transition from single-phase discharge to two-phase discharge through the break. The whole core was uncovered at 208 s after break in Run 8051 (without HPCS actuation). The whole core was also uncovered in Run 804 (with delayed HPCS actuation) and Run 809 (without HPCS actuation) at 210 and 260 s after break, respectively. On the other hand, no dryout occurred in the core in Runs 8061, 807 and 808, in which the HPCS was actuated.

Lower plenum flashing, a typical event for a BWR LOCA, occurred immediately after ADS actuation when the system pressure decreased to a pressure of 6.4 MPa corresponding to the saturation pressure of the lower plenum fluid. Lower plenum flashing caused upward coolant flow in the core and resultant rewetting of fuel rods was observed in the lower core regions.

It was found that the fluid in the feedwater line began to flash at a system pressure of 2.2 MPa, decreasing the depressurization rate of the system and delaying the actuation of LPCI. The inflow from the feedwater line to the pressure vessel due to flashing was detected by a flowmeter in the feedwater line. The ratio of the volume of the water remaining in the feedwater line between the isolation valve and the pressure vessel, to the fluid volume of the ROSA-III pressure vessel is 0.035 which is close to the ratio of 0.033 for a BWR/6. The LPCS was actuated at a system pressure between 2.1 and 2.4 MPa. This contributed to the decrease in the system depressurization rate because it (1) increased steam generation in the core and (2) decreased the steam discharge through the break because of lower quality upstream of the break.

The LPCI was actuated at a system pressure of approximately 1.8 MPa. As LPCI water was injected into the core bypass region, the core was reflooded from the bottom and the quench front proceeded from the bottom to the top. Final quenching of the core was observed after LPCI actuation (between 480 and 540 s after break) in the tests either without HPCS
4.3.2 Thermal-hydraulic Phenomena in Core

The thermal-hydraulic phenomena in the core are very important for the fuel integrity during a LOCA. The liquid level transient estimated from the signals of the conductivity probes is shown in Fig. 2 for Run 8051 (without HPCS actuation). The liquid level outside the core shroud is also shown in the figure for comparison. The core shroud separates the core region from the downcomer in the pressure vessel. The liquid level outside the core shroud began decreasing immediately after the break and the liquid level inside the shroud followed the level fall in the downcomer. The liquid level inside the shroud reached the top of the active core at 120 s after break. There was little difference in the liquid level transients in the average and peak power channels.

Lower plenum flashing began at 231 s after break and the liquid level inside the shroud recovered in the lower part of the core. The liquid level decreased again after 240 s due to less flashing in lower plenum.

The liquid level in the core recovered rapidly after LPCI initiation at 473 s at the rates of 14 and 7 cm/s in the peak and the average power channels, respectively. The whole core was reflooded at 511 s after break.

A typical surface temperature transient observed in Run 8051 (without HPCS actuation) is shown in Fig. 3 for the A33 rod in the center region of the peak power channel. The surface temperature was measured at seven elevations as shown in the figure. Dryout occurred between 129 and 208 s from the top down due to the decrease in the liquid level in the core. The fuel rod surface was rewetted at 231 s at Position 7 (50 mm above the bottom of active core) due to the liquid level rise resulting from lower plenum flashing. The fuel rod surface temperature started to decrease above Position 5 after LPCS initiation at 350 s and top-down quenching started after 386 s from Position 1 to 3 above the midplane of the core. Final bottom-up quenching started at 485 s due to the liquid level rise in the core after LPCI initiation at 473 s. The heater rod surface temperature transient was strongly correlated with the liquid level transient in the core.

The dryout and the quench front transients are compared with the liquid level transient in Fig. 2. Both top-down and bottom-up quenching phenomena were observed after LPCS initiation due to water falling down along the heater rod surface. The falling water was detected by conductivity probes attached to the inner surface of each channel box wall. Top-down quenching by LPCS actuation was not observed in the average power channel D indicating the radial distribution of the water falling into the core. Bottom-up quenching was observed in the peak power channel A and the average power channel D. The final quenching occurred at 538 s after break at Position 1 of the A82 rod in the peripheral region of the peak power channel A and at 523 s at Position 2 of the D27 rod in the corner region of the average power channel D.

The basic trends of the liquid level transients and fuel surface temperatures in Run 804 (with delayed HPCS actuation) and in Run 809 (without HPCS actuation) were the same as the results in Run 8051 (cf. Fig. 4). However, in Run 804 top-down and bottom-up quenching phenomena were observed after HPCS initiation at 236 s after break and the HPCS contributed to the accumulation of coolant in the core. In Run 809, the ADS was actuated at 169 s after break, 60 s earlier than in Run 8051 (without HPCS actuation). The earlier actuation of the ADS resulted in faster depressurization and earlier activation of the LPCS and LPCI, limiting the PCT to a lower value. In Run 809, the LPCS was actuated at 339 s after break, 11 s earlier than in Run 8051 and the LPCI was actuated at 443 s, 30 s earlier.
than in Run 8051. The PCT in Run 809 was 978 K occurring at 450 s after break at the center of the A23 rod, 44 K lower than that in Run 8051.

The PCTs in the six tests are summarized in Fig. 4 and Table 3. In Runs 8061, 807 and 808 in which there was HPCS actuation, no dryout occurred in the core throughout the transient and the fuel surface temperature followed the saturation temperature of the system. The PCT reached 921 K, 1,022 K and 978 K in Run 804 (with delayed HPCS actuation), Run 8051 (without HPCS actuation) and Run 809 (with early actuation of ADS without HPCS actuation), respectively. The observed PCT was still well below the limiting temperature of 1,473 K (1,200 °C) for safety evaluation.

IV.3.3 Comparison with 200% Break Test

Compared in Figs. 5 – 7 are the transients of the lower plenum pressure, liquid level in the core, and peak cladding temperature measured in Runs 8051 and 733. Test conditions for the two tests are the same except for the differences in the break area and in the actuation conditions for the ECCS.

The pressure in both tests shown in Fig. 5 recovered due to the closure of the MSIV (main steam line isolation valve). In Run 8051, the primary system pressure started to decrease rapidly 230 s after break when the automatic depressurization system was actuated and began to discharge steam. In the 200% double-ended break test, Run 733, the system pressure started to decrease rapidly at 12 s after break when the outlet nozzle to the recirculation loop was uncovered in the downcomer and steam was discharged from the break.

In Run 8051, the liquid level started to decrease from the top of core at approximately 120 s and reached the lower tie plate at 208 s (see Fig. 6). Final recovery of the level by LPCI started from the lower tie plate at 485 s and reached the top at 510 s after break. In Run 733, the decrease of the liquid level started at 20 s after break and recovered to the top of the core at 140 s. Accordingly, the length of time during which the core was uncovered and exposed to steam was much longer in Run 8051 than in Run 733. Since the primary system pressure decreased more quickly in Run 733 than in Run 8051, the low pressure injection system was able to inject water into the primary system much earlier in Run 733 than in Run 8051 making the duration of core recovery shorter.

It is important to note that the liquid level in the core recovered temporarily at 18 s after the break in Run 733 and at 231 s in Run 8051, respectively, due to flashing of the fluid in the lower plenum. The fluid in the lower plenum began to flash when the pressure reached the saturation condition due to rapid decrease in the pressure after the liquid level in the downcomer reached the break level in Run 733 and after the ADS was actuated in Run 8051. In a mild and slow transient such as Run 8051, the liquid level in core decreased continuously to the lower plenum, whereas in a rapid, large-break transient, such as Run 733, a separate mixture level was formed in the lower plenum at 30 s after the break while there was still a mixture level in the core. This indicated CCFL occurred at the core inlet orifice in Run 733.

The peak cladding temperature in both tests are shown in Fig. 7. As is clearly shown in the figure, the length of time during which the cladding temperature was above the saturation temperature corresponded exactly to the time during which the core was uncovered. The rate of increase of the cladding temperature was higher in Run 733 since the temperature rise occurred early in the blowdown and the amount of heat generation in the fuel rod was higher in comparison with Run 8051.

It has been found from the above comparisons, that the fundamental phenomena in a 5% break test without HPCS actuation are similar to those in a 200% double-ended break
test without actuation of the LPCS and one LPCI. The core was completely uncovered in both tests but the core was reflooded by ECCS. The primary differences between the two tests were the differences in the mechanism for rapid depressurization and the time span of the transients.

IV.4 Analysis

A post-test analysis of Run 8051 (without HPCS actuation) was performed with the code RELAP4/Mod6\(^3\). Run 8051 resulted in the highest PCT in the 5% small break test series. The objective of the analysis was to study the capability of RELAP4/Mod6 in predicting the thermal-hydraulic phenomena associated with a 5% small break LOCA with boundary conditions measured for the steam flow, feedwater flow and ECCS flows. The break flow was calculated by the code. The primary interest of the analysis was in examining the system pressure, liquid levels in the core and downcomer, and the cladding surface temperatures.

The ROSA-III system was represented by 33 volumes, 50 junctions and 19 heat slabs as shown in Fig. 8. Further description of the calculation model is given in reference (5).

The system pressure calculated by RELAP4/Mod6 is compared with the measured result in Fig. 9. The overall agreement between the calculated and measured pressures is good. However, the calculated pressure is higher than the measured result between 100 and 340 s and becomes lower after 340 s. The vapor generation rate calculated in the core is considered to be in good agreement with the experimental results in the blowdown phase since the calculated and measured heater rod surface temperatures are in good agreement with each other. Therefore, the discrepancy between the calculated and measured pressures shown in Fig. 9 during the blowdown phase is possibly due to less break flow rate calculated in the analysis.

In Figs. 10 and 11, the mixture levels in the core and downcomer calculated by RELAP4/Mod6 are compared with that in the core measured by conductivity probes and the collapsed level in the downcomer measured by differential pressure cells. The calculated mixture level in the core is in good agreement with the experimental results; the decrease in the mixture level due to initiation of blowdown and the swell of the mixture level due to lower plenum flashing after the ADS valve opened are calculated well. The calculated mixture level in the downcomer agrees well with the measured collapsed water level before the initiation of lower plenum flashing; however, the calculated mixture level is higher than the measured collapsed level after the initiation of lower plenum flashing until LPCS actuation because of flashing in the downcomer.

The calculated heater rod surface temperatures are compared with the measured results in Fig. 12. The agreement between the calculated and measured temperatures is good before LPCS actuation except for the lower part of the core (Position 6). The calculated PCT of 930 K occurred at the midplane of the core (Position 4) at the time of LPCS initiation. It is lower than the measured PCT of 1,022 K by 92 K.

There is a strong correlation between the heater rod surface temperatures and the mixture level transient in the core. The rise of the heater rod surface temperature corresponds to the exposure of the heater rod surface to steam both in the test and in the calculation as shown in Figs. 10 and 12. The above observations for the blowdown phase indicate that the heat transfer correlations are adequately selected in the code for the blowdown calculation.

In the reflooding phase, the heater rod surface in the analysis was not quenched from the top by LPCS water but the temperatures decreased gradually after initiation of LPCS. The temperature decrease in heater rod surface was accelerated after the initiation of LPCI and the heater rod surface temperatures returned to a value close to the saturation tempera-
ture of the system pressure. The heater rod surface was not quenched although the core was reflooded. One possible reason for the gradual decrease in heater rod surface temperature after reflooding may be due to the neglect of axial heat conduction in the RELAP4/Mod6 analysis. The reflooding model of RELAP4/Mod6 was not used in the present analysis because its application is limited to PWR systems.

The present analysis has confirmed that the core reflooding and decrease of the cladding surface temperature close to the saturation temperature can be calculated without any special reflood model if a sufficiently small time step is used.

The rapid decrease in the measured density at the vessel side of the break corresponded to the recovery of the recirculation outlet nozzle to steam in the downcomer as the level dropped in the downcomer. The cumulative break flow rate can be checked out by the timing of the exposure. The density measured at the vessel side of break began to decrease rapidly at 216 s after the break, whereas the density calculated was that of liquid until flashing occurred in the downcomer at 231 s after the break. Therefore, it is clear that the calculated break flow rate is less than the experimental results. Less break flow resulted in a higher system pressure, a larger amount of residual water in the pressure vessel and a slower rate of decrease in the downcomer level (see Figs. 9 and 11).

IV.5 Conclusions

The following conclusions were obtained for a 5% break LOCA from the experimental results:

1. The fuel surface dries out and the temperature starts to rise as the mixture level decreases due to the break flow, and the fuel surface quenches after the core is reflooded following LPC1 actuation. The peak cladding temperature depends on the times of dryout and reflooding.

2. With HPCS actuation, the fuel rods are covered by the two-phase mixture throughout the transient, regardless of LPCS and LPC1 conditions, and the fuel surface temperature follows the system saturation temperature.

3. Delay in HPCS actuation can result in core recovery. Without HPCS actuation, the whole core is uncovered to the steam environment; however, the fuel rods are quenched by the LPCS and LPC1 after ADS actuation. Therefore, early actuation of the ADS IS important to limit the peak cladding temperature to a low value in a 5% small break LOCA with HPCS failure.

4. The fundamental phenomena in a 5% break test without HPCS actuation are similar to those in a 200% double-ended break test without actuation of the LPCS and one LPC1. The whole core is uncovered to the steam environment in both tests and reflooded finally by ECCS.

The following conclusions have been obtained from the analysis of the 5% break test Run 8051 (without HPCS actuation) with RELAP4/Mod6:

1. The calculated system pressure after the main steam isolation valve closure is higher than the measured data possibly due to less break flow in the calculation before ADS actuation.

2. The trend of the mixture level transients calculated using Wilson's bubble rise velocity correlation agrees well with the experimental results inside and outside the shroud.

3. The calculated heater rod surface temperatures are in good agreement with the experimental results in the blowdown phase.

It has been confirmed from the results of the ROSA-III 5% small break test series that
no unexpected phenomena occurred in the small break LOCA except for feedwater line flashing and the resultant delay in LPCI actuation.

References


Table 1 Summary of test conditions

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† Recirculation pump, ‡ Diesel generator
### Table 2  Sequence of major events

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<tr>
<td>Feedwater line closure</td>
<td>2.0</td>
<td>2.0</td>
<td>2.0</td>
</tr>
<tr>
<td>Steam line closure</td>
<td>26.0</td>
<td>26.4</td>
<td>28.0</td>
</tr>
<tr>
<td>HPCS initiation</td>
<td>236</td>
<td>50.0</td>
<td>50.0</td>
</tr>
<tr>
<td>Jet pump suction uncovering</td>
<td>101</td>
<td>129</td>
<td></td>
</tr>
<tr>
<td>First dryout in core</td>
<td>132</td>
<td>129</td>
<td></td>
</tr>
<tr>
<td>Recirc. pump suction uncovering</td>
<td>185</td>
<td>212</td>
<td></td>
</tr>
<tr>
<td>ADS actuation</td>
<td>229</td>
<td>228</td>
<td>329</td>
</tr>
<tr>
<td>Lower plenum flashing</td>
<td>233</td>
<td>231</td>
<td>330</td>
</tr>
<tr>
<td>LPCI initiation</td>
<td>449</td>
<td>473</td>
<td>506</td>
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<tr>
<td>Final quench of core</td>
<td>481</td>
<td>538</td>
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### Table 3  Comparison of peak cladding temperatures

<table>
<thead>
<tr>
<th>Run</th>
<th>804</th>
<th>8051</th>
<th>8061</th>
<th>807</th>
<th>808</th>
<th>809</th>
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</thead>
<tbody>
<tr>
<td>Single failure</td>
<td>LPCS-DG&lt;sup&gt;†1&lt;/sup&gt;</td>
<td>HPCS</td>
<td>LPCI-DG&lt;sup&gt;†2&lt;/sup&gt;</td>
<td>–</td>
<td>LPCS-DG</td>
<td>HPCS</td>
</tr>
<tr>
<td>Comment</td>
<td>HPCS act,&lt;sup&gt;†3&lt;/sup&gt;</td>
<td>186 s later</td>
<td>Full ECCS</td>
<td></td>
<td>ADS act. 60 s earlier</td>
<td></td>
</tr>
<tr>
<td>PCT (K)</td>
<td>912</td>
<td>1,022</td>
<td>569</td>
<td>569</td>
<td>569</td>
<td>978</td>
</tr>
<tr>
<td>Time (s)</td>
<td>433</td>
<td>364</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>450</td>
</tr>
<tr>
<td>Position&lt;sup&gt;†4&lt;/sup&gt;</td>
<td>4</td>
<td>2</td>
<td>–</td>
<td>–</td>
<td>–</td>
<td>4</td>
</tr>
<tr>
<td>Rod&lt;sup&gt;†5&lt;/sup&gt;</td>
<td>A34</td>
<td>A34</td>
<td>–</td>
<td>–</td>
<td>–</td>
<td>A23</td>
</tr>
</tbody>
</table>

<sup>†1</sup> Single failure of LPCS diesel generator resulting in the failure of LPCS and one out of three LPCI systems.

<sup>†2</sup> Single failure of LPCI diesel generator resulting in the failure of two out of three LPCI systems.

<sup>†3</sup> Actuation time compared with the standard test.

<sup>†4</sup> Position 1 = 50 mm below the top of active core, Position 2 = 588 mm above the center, Position 3 = 353 mm above the center, Position 4 = center, Position 5 = 353 mm below the center, Position 6 = 588 mm below the center, Position 7 = 50 mm above the bottom.

<sup>†5</sup> see Fig. 2.4(b)
Fig. 1 Comparison of system pressures.

Fig. 2 Liquid level in Run 8051.

Fig. 3 Fuel surface temperatures in Run 8051.
Fig. 4  Comparison of peak cladding temperatures.

Fig. 5  Lower plenum pressures in Runs 733 and 8051.

Fig. 6  Liquid levels in Runs 733 and 8051.
Fig. 7  Peak cladding temperatures in Runs 733 and 8051.

Fig. 8  ROSA-III node and junction representation for RELAP4/Mod6 analysis.

Fig. 9  System pressure.
Fig. 10  Liquid level inside shroud.

Fig. 11  Liquid level outside shroud.

Fig. 12  Cladding surface temperatures.
Appendix V  Publications in ROSA-III Experimental Program


44) Tasaka, K., et al.: The LOCA/ECC system effects tests at ROSA-III changing the break
46) Tasaka, K., et al. : BWR LOCA/ECCS integral test at ROSA-III, (Recirculation pump discharge line and steam line breaks), 10th Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, USA (Oct. 1982).
51) Tasaka, K., et al. : Steam line break, jet pump drive line break and natural circulation tests in ROSA-III program for BWR LOCA/ECCS integral tests, 11th Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, USA (Oct. 1983).
65) Iriko, M., Kukita, Y. and Tasaka, K. : Assessment of the THYDE-B1/MOD0 code


87) Tasaka, K., et al.: Comparisons of ROSA-III and FIST BWR loss-of-coolant accident