FINAL SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
TOPICAL REPORT MUAP-07013-P, REVISION 2
"SMALL BREAK LOCA METHODOLOGY FOR US-APWR"
MITSUBISHI HEAVY INDUSTRIES, Ltd
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1.0 INTRODUCTION AND BACKGROUND

Mitsubishi Heavy Industries, Ltd. (MHI) submitted Topical Report MUAP-07013-P (R0), titled "Small Break LOCA Methodology for US-APWR," in letter UAP-HF-07092 [MHI01], dated July 20, 2007. The MHI methodology uses the M-RELAP5 computer code to carry out the emergency core cooling system (ECCS) performance analysis for small-break loss-of-coolant accidents (SBLOCAs). The regulatory basis for the LOCA evaluation is specified in 10 CFR §50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," [NRC01]. 10 CFR 50, Appendix K, "ECCS Evaluation Models," specifies required and acceptable features of ECCS evaluation models (EM) [NRC02].

The purpose of submitting Topical Report MUAP-07013-P (R0) during the United States Advanced Pressurized Water Reactor (US-APWR) pre-application phase was to provide information to the NRC to facilitate efficient and timely review of the accident analyses to be provided in the Design Control Document (DCD) as part of the Design Certification License Application.

MHI also provided SBLOCA sensitivity studies, in Technical Report MUAP-07025-P (R0), "Small Break LOCA Sensitivity Analyses for US-APWR" [MHI02]. The results of the break spectrum sensitivity calculations identified the limiting break conditions including break location, break size, and break orientation. Breaks in the cold leg piping were determined to be limiting for peak clad temperature (PCT). During the loop-seal clearing phase the PCT occurred for a 7.5-inch cold leg break, while during the boiloff phase the PCT occurred for the 1.0 ft^2 cold leg break. The sensitivity calculations also showed that noding near the break and the reactor coolant system (RCS) noding were appropriate for the licensing analyses of SBLOCAs in the US-APWR. The time step size was sufficiently small for code solution convergence. The analyses results also demonstrated that the assumptions for single failures and for Loss-of-Offsite Power (LOOP) were satisfactorily selected.

The NRC staff, hereinafter referred to as the staff, reviewed Topical Report MUAP-07013-P (R0) and requested additional information related to the use of the M-RELAP5 code and its application for the US-APWR SBLOCA analyses performed to demonstrate compliance with applicable NRC rules and regulations for the ECCS.

MHI responded to the first request for additional information (RAI) [NRC03], in MHI letter UAP-HF-09002 [MHI03] and MHI letter UAP-HF-09041 [MHI04]. MHI responded to the second RAI [NRC04], in MHI letter UAP-HF-09362 [MHI05] and MHI letter UAP-HF-09471 [MHI06]. MHI responded to the third RAI [NRC05], in MHI letter UAP-HF-09492 [MHI07] and MHI letter UAP-HF-09512 [MHI08]. MHI responded to the fourth RAI [NRC06], in MHI letter UAP-HF-09559 [MHI09] and letter UAP-HF-10003 [MHI10]. MHI responded to the fifth RAI [NRC07], in MHI letter UAP-HF-10059 [MHI11] and MHI letter UAP-HF-10074 [MHI12].

MHI provided a report titled "Scaling Analysis for US-APWR Small Break LOCAs," in MHI letters UAP-HF-09472 [MHI13], Part 1 of the report, UAP-HF-09541 [MHI14], Part 2 of the report and UAP-HF-09568 [MHI15], the full report. The report provided an evaluation of the scalability of the experimental test facilities used for the M-RELAP5 code assessment for application to US-APWR SBLOCA analyses. In addition, the capabilities of the code governing equations, models and correlations were also investigated in the framework of this study. MHI provided responses to a seventh RAI [NRC08] in MHI letter UAP-HF-10151 [MHI16]. MHI provided a revised

scaling analysis for US-APWR SBLOCAs in UAP-HF-10289 [MHI17], which incorporated the responses to the RAI.

MHI provided a report titled "M-RELAP5 Additional Code Assessment Using LOFT/L3-1 and Semiscale/S-LH-1 Test Data," in MHI letter UAP-HF-09567 [MHI18]. In this report the M-RELAP5 code was assessed based on the requirement from the Three Mile Island (TMI) action plan [NRC09] Item II.K.3.30, "Revised small-break LOCA analysis." The action plan states that the computer codes used for safety analyses shall be validated using simulated SBLOCA integral experimental test data, specifically, the loss-of-fluid test (LOFT) and Semiscale test facilities. In conformance with this requirement, the LOFT Test L3-1 and Semiscale Test S-LH-1 were selected for an additional code assessment to demonstrate the M-RELAP5 capability for US-APWR SBLOCA analyses. The report described the M-RELAP5 code validation results using these experimental data. MHI provided responses to a sixth RAI [NRC10] in MHI letter UAP-HF-10113 [MHI19] and UAP-HF-10137 [MHI20].

MHI provided Revision 1 to Topical Report MUAP-07013-P [MHI21]. This revision included additional integral test facility analyses to support the use of M-RELAP5 for the US-APWR SBLOCA ECCS evaluation, including Section 8.2.2, "ROSA/LSTF Small Break (10 percent) LOCA test (SB-CL-09)," and Section 8.2.3, "ROSA/LSTF Small Break (17 percent) LOCA test (IB-CL-02)." The LOFT assessment was added in Section 8.2.4, "LOFT Small Break (2.5 percent) LOCA test (L3-1)." The Semiscale assessment was added in Section 8.2.5, "Semiscale Small Break (5 percent) LOCA test (S-LH-1)."

MHI provided Revision 2 to Topical Report MUAP-07013-P [MHI22]. This revision forms the bases for the M-RELAP5 (M1.6) computer program and its applicability to the evaluation of SBLOCAs in the US-APWR for licensing analyses to demonstrate conformance with the NRC rules and regulations. A companion revision to the SBLOCA sensitivity studies, Technical Report MUAP-07025-P (R2), was provided in Reference MHI23. Technical Report MUAP-07025-P (R3) was provided in Reference MHI24, to fulfill a commitment from a January 31, 2011, and February 1, 2011 staff meeting with MHI regarding the US-APWR advanced accumulator and had no impact on this review. This revision added sensitivities studies for the advanced accumulator flow rate.

The complete technical evaluation of Topical Report MUAP-07013-P is provided in Reference ISL01. The technical evaluation report addresses all RAIs and provides additional technical details regarding the M-RELAP5 code. This safety evaluation (SE) addresses those RAIs that were used to support conclusions regarding the acceptability of M-RELAP5 for US-APWR SBLOCA analyses.

This SE is limited to the use of M-RELAP5 for US-APWR SBLOCA analyses performed to demonstrate compliance with applicable NRC rules and regulations for the ECCS.

2.0 REGULATORY EVALUATION

2.1 REGULATORY REQUIREMENTS

The acceptance criteria for a LOCA are based on meeting the relevant requirements of the following Commission regulations:

(1) 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors” [NRC01] as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

(2) General Design Criterion [NRC10] 35, “Emergency Core Cooling,” as it relates to demonstrating that the ECCS will provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling will be prevented, and (2) clad metal-water reaction will be limited to negligible amounts. The analyses should reflect that the ECCS has suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities available such that the safety functions could be accomplished assuming a single failure. In addition, consideration should be given to the availability of onsite power (assuming offsite electric power is not available with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available).

The requirements specified in 10 CFR 50.46 provide an acceptable and conservative means for the calculation of the consequences of LOCAs resulting from a spectrum of pipe break sizes and locations. These requirements have been subjected to careful review and experimental verification. Appendix K to 10 CFR Part 50, “ECCS Evaluation Models” [NRC02] provides guidance and requirements for EMs needed to demonstrate compliance with the acceptance criteria.

The US-APWR SBLOCA analyses are performed to demonstrate that the following limits, set forth in 10 CFR 50.46, are met:

(b)(1) Peak cladding temperature.

PCT: The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

(b)(2) Maximum cladding oxidation.

Maximum cladding oxidation: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(b)(3) Maximum hydrogen generation.

Maximum hydrogen generation: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding

cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(b)(4) Coolable geometry.

The acceptance criterion requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b) (1) and (b) (2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed.

(b)(5) Long-term cooling.

After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The acceptance criterion requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core.

2.2 Regulatory Guidance

Guidance for the review of LOCA analyses is provided in the Standard Review Plan (SRP) 15.6.5, “Loss-Of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” [NRC11].

The SRP 15.6.5 review of the analysis of the spectrum of postulated LOCAs is closely associated with the review of the ECCS, as described in SRP Section 6.3, “Emergency Core Cooling System” [NRC11]. A portion of the review effort described in SRP 15.6.5 and in SRP Section 6.3 evaluates whether the entire break spectrum (break size and location) has been addressed; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance have been provided. For postulated break sizes and locations, the review includes the postulated initial reactor core and reactor system conditions; the postulated sequence of events including time delays prior to and after emergency power actuation; the calculation of the power, pressure, flow and temperature transients; the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events; and operator actions required to mitigate the consequences of the accident.

The MHI SBLOCA methodology is based on satisfying the criteria specified in Appendix K to 10 CFR 50.

Regulatory Guide (RG) 1.203, “Transient and Accident Analysis Methods,” [NRC12] provides guidance for the review of the methods to be used to analyze the plant response to transients and accidents. The requirements regarding applications for construction permits and/or licenses to operate a facility are:

(1) Safety analysis reports must analyze the design and performance of structures, systems, and components (SSCs), and their adequacy for the prevention of accidents and mitigation of the consequences of accidents.

(2) Analysis and evaluation of the ECCS cooling performance following postulated loss-of-coolant accidents (LOCAs) must be performed in accordance with the requirements of 10 CFR 50.46.

(3) The technical specifications (TS) for the facility must be based on the safety analysis and prepared in accordance with the requirements of 10 CFR 50.36.

Section 15.0.2, “Review of Transient and Accident Analysis Method,” of the SRP [NRC11] provides guidance to the NRC reviewers of transient and accident analysis methods.

EM Concept

The EM concept establishes the basis for methods used to analyze a particular event or class of events. This concept is described in 10 CFR 50.46 for LOCA analysis, but can be generalized to all analyzed events described in the SRP.

An EM is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) Procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation).

(2) Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used.

(3) All other information needed to specify the calculational procedure.

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

Specifically, the following six basic principles have been identified as important to follow in the process of developing and assessing an EM:

(1) Determine requirements for the EM.

(2) Develop an assessment base consistent with the determined requirements.

(3) Develop the EM.

(4) Assess the adequacy of the EM.

(5) Follow an appropriate quality assurance protocol during the EM development and assessment process (EMDAP).

(6) Provide comprehensive, accurate, up-to-date documentation.

In addition, as part of the code assessment, a scaling analysis that identifies important non-dimensional parameters related to geometry and key phenomena, must be performed. Scaling distortions and their impact on the code assessment must be identified and evaluated.

3.0 SUMMARY OF US-APWR SBLOCA APPLICATION

3.1 ROADMAP FOR THE DEVELOPMENT OF THE M-RELAP5 EVALUATION MODEL

Section 1.0 of Topical Report MUAP-07013-P provided the roadmap for the development and assessment of the M-RELAP5 EM.

MHI developed the M-RELAP5 computer code for the US-APWR SBLOCA analysis from the RELAP5-3D computer code based on two important principles:

(1) MHI determined this to be the most straightforward way to satisfy the basic requirements for the development and assessment of a SBLOCA EM as described in RG 1.203 [NRC12]. RELAP5 is a mature code that incorporates the modeling approaches and specific models required to model a wide range of transients in plant designs similar to the US-APWR. RELAP5-3D has also been directly applied to most of the experiments applicable to SBLOCAs in pressurized-water reactors (PWRs). The development of RELAP5-3D has followed quality assurance (QA) standards with independent peer review [TEC01] as a fundamental part of its development history.

(2) RELAP5-3D is the culmination of a long series of RELAP5 versions developed at the Idaho National Laboratory. Many of the staff involved with current Code development and application have been associated with the Code over much of its development history. RELAP5-3D models and correlations are based on the widely accepted and tested RELAP5/MOD3.2 models and correlations first released in the NRC versions of RELAP5. Many of the current user guidelines have been prepared by staff members involved in the development and validation of the Code.

3.2 COMPLIANCE WITH 10 CFR 50.46

Section 2.0 of Topical Report MUAP-07013-P describes the approach taken by MHI to demonstrate compliance with 10 CFR 50.46.

The report describes MHI's analysis methodology and evaluation of the ECCS cooling performance for design-basis SBLOCAs in the US-APWR. These analyses were performed in accordance with the requirements regarding applications specified in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.34, "Contents of Applications; Technical Information."

The purpose of the analysis was to demonstrate the evaluation and performance of the ECCS for the design-basis SBLOCAs in the US-APWR in accordance with the requirements specified in 10 CFR Part 50 Section 50.34, and the acceptance criteria specified in 10 CFR Part 50 Section 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light-Water

Nuclear Power Reactors." These figures of merit are provided in Section 2.1, "Regulatory Requirements," of this SE.

Topical Report MUAP-07013-P was prepared to conform to the approach shown in RG 1.203, "Transient and Accident Analysis Methods." This approach is summarized in Section 2.2, "Regulatory Guidance," of this SE.

The SBLOCA events are categorized as one of the postulated design-basis accidents that are specified in SRP 15.0. SRP 15.6.5 states in the "AREAS OF REVIEW" section that a spectrum of both LBLOCAs and SBLOCAs are to be evaluated and the limiting breaks are to be identified through sufficient analyses to determine the worst break PCT, the worst local clad oxidation, and the highest core-wide clad oxidation percentage. Moreover, the SRP states that for the evaluation of the ECCS, the EM must comply with acceptance criteria for ECCS given in 10 CFR 50.46.

The postulated SBLOCA is defined by MHI as a break in the reactor coolant pressure boundary that results in a loss-of-coolant at a rate in excess of the capability of the normal reactor coolant makeup system and is equal to or less than 1.0 ft^2.

The US-APWR SBLOCA event is divided into five periods that characterize the fluid transient behavior in the RCS. They are: Blowdown, Natural Circulation, Loop Seal Clearance, Boiloff, and Core Recovery. The duration of each period is break-size dependent. The above classification was used to identify and rank various phenomena to develop a Phenomena Identification and Ranking Table (PIRT), which is discussed in Section 4.0 of Topical Report MUAP-07013-P

3.3 SYSTEMS, COMPONENTS, PHASES, GEOMETRIES, FIELD EQUATIONS, AND PROCESSES MODELING

Section 3.0 in Topical Report MUAP-07013-P describes the US-APWR systems and components that need to be modeled in the M-RELAP5 Appendix K EM. This section describes the thermodynamic phases occurring during a SBLOCA that need to be modeled in the EM, as well as the system and component geometries that also need to be modeled in the EM. The thermodynamic field equations and processes are also described.

During a SBLOCA, the thermal-hydraulic transient is of longer duration than for a LBLOCA since the rate of discharged flow and energy is relatively small compared to a LBLOCA. To ensure the appropriateness of the thermal-hydraulic plant response, the reactor system model included the reactor core, the RCS, the ECCS, and the secondary system.

The M-RELAP5 representation of the US-APWR uses one-dimensional (1D) modeling to construct the integrated reactor system model by interconnecting the 1D control volumes of the reactor system with flow paths representing the flow paths that exist in the US-APWR.

The 1D thermal-hydraulic code, M-RELAP5, used for the SBLOCA analysis contains well-established two-phase flow regimes. The transient behavior of the system is analyzed using governing equations of mass, energy and momentum, as modeled in the Code. The Code allows for multi-node modeling to represent the spatial depiction of the reactor core, and also includes the following models specific to SBLOCA transients:

- Critical flow correlations.

- Heat transfer between the core and metal structures and fluid flow.

- Response of components including pump coastdown, valve opening/closing, and accumulator discharging behavior.

- Signals to actuate or trip equipment.

3.3.1 Systems and Components

The general system configuration of US-APWR is equivalent to that of the Westinghouse-designed four-loop PWR, with the thermal-hydraulic volume, flow area, and diameter of reactor components and their piping sized to accommodate the larger thermal output of US-APWR. The US-APWR is rated at 4,451 MWt.

The US-APWR systems that must be modeled and analyzed include:

(1) Primary System (reactor and core, RCS, ECCS).

(2) Secondary System (main steam system, main feedwater system, emergency feedwater system).

(3) Containment Vessel.

The reactor primary and steam generator (SG) secondary systems are modeled in the SBLOCA calculations. Primary system modeling includes the reactor internals and vessel, the SGs, the reactor coolant pumps (RCPs), the pressurizer, the reactor coolant piping and pressurizer surge line, the accumulators and the high-head safety injection (SI) system. Secondary system modeling includes the SG secondary side, and the main feedwater, main steam and emergency feedwater lines, their isolation valves, and safety and relief valves.

3.3.1.1 Primary System Components

The primary system contains the reactor and core, the RCS and the ECCS

Reactor and Core

The US-APWR fuel assembly utilizes a 17x17 array of 264 fuel rods, 24 control rod guide thimbles and one in-core instrumentation guide tube. The fuel rod and thimble components are bundled by grid spacers. The fuel design uses 11 grid spacers that span the 13.78-ft (4.2 m) active fuel length. The grid-to-grid distance for the US-APWR design is basically the same as that for the 12-ft (3.66 m) Mitsubishi fuel with a nine grid spacer design, thus ensuring a similar resistance to failures due to fretting wear, and the same coolant mixing and departure from nucleate boiling (DNB) performance as the 12-ft (3.66 m) fuel design.

The reactor internals consist of two major assemblies, the lower core support assembly and the upper core support assembly. These support the core, maintain fuel assembly alignment, limit fuel assembly movement, and maintain alignment between fuel assemblies and control rods. These structures also direct the coolant flowing through the fuel assemblies, transmit the loads

from the core to the reactor vessel (RV), provide radiation shielding of the RV, and guide the in-core instrumentation.

Reactor Coolant System

The RCS consists of the RV, the SGs, the RCPs, the pressurizer, and the reactor coolant pipes and valves.

The RV contains the fuel assemblies and RV internals, including the core support structures, control rods, neutron reflector and other structures associated with the core. The RV has four inlet nozzles, four outlet nozzles, and four SI nozzles, which are located between the upper RV flange and the top of the core. The SG is a vertical shell U-tube evaporator with integral moisture separating equipment.

The RCPs are vertical single-stage centrifugal pumps of similar design to a Westinghouse 93A pump, which is used in four-loop PWRs, and is driven by three-phase induction motors. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The pump suction is located at the bottom of the pump, and the discharge on the side of the pump. The US-APWR has an automatic RCP trip, with a three second delay, on an ECCS SI signal generated from low pressurizer pressure or high containment pressure, as required by TMI action item II.K.3.5, "Automatic RCP Trip during a LOCA."

The pressurizer functions to control the RCS pressure and to accommodate changes in the coolant volume. The pressurizer is a vertical vessel with hemispherical top and bottom heads. Electrical immersion-type heaters are installed vertically through the bottom head of the vessel. The spray nozzle and relief line connections to the relief and safety valves are located on the top head of the vessel.

The reactor coolant pipe network consists of the pipes connecting the reactor pressure vessel, SGs, RCPs, and pressurizer.

3.3.1.2 Emergency Core Cooling System

The ECCS injects borated water into the reactor coolant system following a postulated accident and performs the following functions:

- Following a LOCA, the ECCS cools the reactor core, prevents the fuel and fuel cladding from serious damage, and limits the zirconium-water reaction of the fuel cladding to a very small amount.

- Following a main steam line break (MSLB), the ECCS provides negative reactivity to shut down the reactor.

- In the event that the normal chemical and volume control system (CVCS) letdown and boration capability is lost, the ECCS provides emergency letdown and boration of the RCS.

The ECCS design is based on the following requirements:

(1) In combination with control rod insertion, the ECCS is designed to shut down and cool the reactor during the following accidents:

- LBLOCA and SBLOCA of the primary piping.

- Control rod ejection.

- Main steamline break.

- SG tube rupture

(2) The ECCS is designed with sufficient redundancy (four trains) to accomplish the specified safety functions assuming a single failure of an active component following an accident with one train out of service for maintenance, or a single failure of an active component or passive component for the long term following an accident with one train out of service.

(3) The ECCS is automatically initiated by a SI signal.

(4) The emergency electrical power to the essential components is provided so that the design functions can be maintained during a LOOP.

The ECCS includes the accumulator system, the high-head SI system, and the emergency letdown system. The accumulator system and high-head SI system are included in the US-APWR SBLOCA EM.

Accumulator System

The accumulator system [MHI25], which is a passive safety component, consists of four accumulators, and the associated valves and piping, one for each RCS loop. The system is connected to the cold legs of the reactor coolant piping and injects borated water when the RCS pressure falls below the accumulator operating pressure. Pressurized nitrogen gas forces borated water from the tanks into the RCS. The accumulator performs the large flow injection to refill the RV, and then provides a smaller injection flow during core reflooding in association with the high-head SI pumps. The high-head SI system provides long term core cooling.

High-Head Injection System

The high-head injection system (HHIS), which is an active safety component, consists of four independent trains, each containing a SI pump and the associated valves and piping. The SI coolant is directly injected into the downcomer using direct vessel injection (DVI). The SI pumps start automatically upon receipt of the SI signal. One of four independent safety electrical buses is available to each SI pump. The SI pumps are aligned to take suction from the refueling water storage pit (RWSP) and to deliver borated water to the SI nozzles on the RV. Two SI trains are capable of meeting the design cooling function for a large or small break LOCA. This capability ensures adequate ECC delivery in the case where it is assumed that there is a single failure in one train and a second train is out of service for maintenance.

The high-head SI system provides long term core cooling for the core. The SI pumps start automatically upon receipt of the SI signal. One of four independent safety electrical buses is available to each SI pump.

The RWSP, in the containment, provides a continuous borated water source for the SI pumps. This configuration eliminates the need for realignment from the refueling water storage tank to the containment sump, which is employed in existing PWR plants.

3.3.1.3 Secondary System Components

The secondary system consists of the main steam system, the main feedwater system, the emergency feedwater system, and the power conversion system.

Main Steam System Components

The main steam system includes the main steam pipes from the SG outlets to the turbine inlet steam chests and equipment and piping connected to the main steam pipes. The main steam relief and safety valves, which prevent excessive steam pressure and maintain cooling of the RCS if the turbine bypass is not available, are installed upstream of the main steam isolation valve. The total capacity of the main steam safety valves exceeds 100 percent of the rated main steam flow rate. Branch pipes for driving the turbine-driven emergency feedwater pumps are connected upstream of the main steam isolation valves. The secondary sides of the SGs up to the main steam isolation valves are included in the US-APWR SBLOCA EM.

Main Feedwater System Components

The main feedwater system supplies the SGs with heated feedwater in a closed steam cycle using regenerative feedwater heating. The system is composed of the condensate subsystem, the feedwater subsystem, and a portion of the SG feedwater piping. The feedwater control valves, the feedwater bypass control valves, the SG water filling control valves, and the feedwater isolation valves are installed on the feedwater lines. The feedwater isolation valves to the secondary sides of the SGs are included in the US-APWR SBLOCA EM.

Emergency Feedwater System Components

The emergency feedwater system (EFWS) consists of two motor-driven pumps, two steam turbine-driven pumps, two emergency feedwater pits, and associated piping and valves. The four emergency feedwater pumps take suction from two emergency feedwater pits. The EFWS removes reactor decay heat and RCS residual heat through the SGs following transient conditions or postulated accidents.

3.3.1.4 Containment Vessel

The containment vessel is designed to completely enclose the reactor and RCS and to ensure that essentially no leakage of radioactive materials to the environment would result even if a major failure of the RCS were to occur. The containment vessel is a pre-stressed, post-tensioned concrete structure with an inside steel lining. The containment vessel is designed to contain the energy and radioactive materials that could result from a postulated LOCA. In the US-APWR SBLOCA EM, an atmospheric condition inside the containment vessel is assumed as the boundary conditions for the break back-pressure in the M-RELAP5 US-APWR SBLOCA analyses.

3.3.2 Transient Phases

Sensitivity studies performed in Technical Report MUAP-07025-P by MHI have shown a bottom-oriented cold leg break as the limiting break location in terms of core cooling. The behavior of the RCS is characterized by single-phase forced and natural circulation, followed by two-phase combined forced and natural circulation during the depressurization and SI phases of the accident. The secondary side is also characterized by a combination of single-phase and two-phase natural convection when the secondary side is isolated from the turbine generator and steam is released through the secondary side relief valves. The primary system may have non-condensable gases present during part of the accident from the injection of nitrogen once the accumulators empty.

The transition from single-phase forced convection to two-phase convection occurs during the blowdown (BLD) phase of the accident in the US-APWR. As the RCS depressurizes, the pressure in the pressurizer reaches the "pressurizer pressure low" set point (1860 psia (12.82 MPa)), the reactor is tripped and the SG secondary side is isolated. As a result, the SG secondary side pressure rises to the safety valve set point and secondary side steam is released through the safety valves.

When the pressurizer reaches the "pressurizer low-low pressure" set point (1760 psia (12.13 MPa)), a SI signal is generated and ECCS SI will be initiated after a specified delay time. The ECCS SI is comprised of the four train DVI high pressure pumped system and the four advanced accumulators, which inject into the cold leg. The accumulators will inject water once the primary system pressure falls below the accumulator operating pressure (600 psia (4.14 MPa).

The primary system liquid remains single-phase for much of the BLD period, with phase separation first occurring in the upper head, upper plenum and hot legs. The break flow remains liquid during this period. As the pressure continues to drop, the primary system pressure will reach the saturation point, ending the subcooled phase of the BLD period. The saturated BLD will continue as the primary side pressure approaches the secondary side pressure.

As the BLD phase ends, a period of two-phase natural circulation will occur. Two-phase natural circulation will continue as long as condensation occurs in the SG tubes. At some point, natural circulation will stop as vapor builds up in the SG tubes and blocks the circulating flow.

The next period is characterized by the clearance of the cold leg loop seal with the possibility of partial core uncovery due to the static pressure imbalance in the primary system. After the loop seal clears, the water level in the vessel will rise. The remaining phases will depend on the size of the break and the capacity of the water injection systems. If the break flows exceed the capacity of the water injection systems, the vessel water level will again start to decrease. Otherwise the vessel water levels will increase and the transient will be terminated.

The injection of non-condensable gases from the accumulators as they empty can occur, although the specific timing and quantities of non-condensable gases present at any time will depend on the transient. The accumulators must empty before the non-condensable gas (nitrogen) is injected into the RCS. It is expected that the core will be fully recovered during a SBLOCA in the US-APWR before the accumulators empty and the accumulator nitrogen gas will not enter the RCS during the period of interest.

Both single-phase and two-phase flow behavior including the influence of the pressure differences, heat transfer, and co-current and counter-current flow need to be modeled for a SBLOCA. Specific models that represent each of the systems and components of the US-APWR are modeled as well as models specific to PWR fuel assemblies that are used to describe the flow and heat transfer in the core region. The core region can experience both single-phase and two-phase flows as well as co- and counter-current flows, including the influence of water drainage from the upper plenum region. Models that include the specific features of the other RV internals are also used. Like the core region, these components include the downcomer, upper plenum, and lower plenum. During the transient, these components can also experience both single- and two-phase mixtures with the presence of non-condensable gases.

Single- and two-phase co- and counter-current flow is also modeled in the balance of the primary system and secondary system components. For example, the water held up in the SG due to flooding is modeled.

3.3.3 Geometries

All components of the primary system and portions of the secondary system are modeled for the SBLOCA. Where appropriate, portions of the ECCS and containment are included in the analysis. The specific components of the ECCS and containment systems are activated using appropriate time dependent boundary conditions. These systems, and the structures associated with them, are modeled using 1D flow networks and heat structures.

The geometries of the flow paths and structures that make up the US-APWR primary system and components are modeled so that flow rates, pressure differences, and heat transfer can be calculated. Representative fuel assembly geometries are modeled in the core region including the fuel rod dimensions, fuel assembly pitch, and other physical characteristics of the fuel assemblies. The physical characteristics of the other reactor internal structures are modeled including (a) primary flow areas, (b) leakage paths such the paths between the upper plenum and downcomer, and (c) structural surface areas and volumes to insure the proper heat storage in these structures. The flow areas, orientation, and structural surface areas and thicknesses are modeled in the balance of the primary system and components to insure acceptable flow and heat transfer calculations.

The geometries of the secondary system flow paths and structures that are important to SBLOCA conditions are modeled to insure the adequacy of the calculations of the secondary side heat transfer and flow conditions.

3.3.4 Field Equations

A non-equilibrium, separated two-phase flow model is used in M-RELAP5 to model the SBLOCA in the US-APWR. The M-RELAP5 model includes the effect of non-condensable gases. The M-RELAP5 one-dimensional formulation is used to model the primary and secondary systems.

The basic field equations for the two-fluid non-equilibrium model in M-RELAP5 consist of two phasic continuity equations, two phasic momentum equations, and two phasic energy equations. The phase change between the phases is calculated from the interfacial and wall heat and mass transfer models. The basic two-phase single-component model is extended to

include a noncondensable component in the vapor/gas phase. State relationship equations and constitutive equations make up closure relations for the system of basic field equations.

3.3.5 Processes

All of the processes important to the analysis of US-APWR SBLOCA are considered along with those processes that are useful for the purposes of the analysis, such as the implementation of control system responses. The thermal hydraulic response of the primary and secondary system, heat transfer within system structures and components, power generation associated with fission heating, decay heat, and oxidation of the fuel rod cladding, and the important features of the reactor control systems are modeled as needed for a specific transient.

The deformation and rupture of the fuel rod cladding and its impact on flow within the core is modeled. However, no significant clad deformation or rupture is predicted to occur for SBLOCA events in the US-APWR.

Specifically, the processes to be considered include:

- Single-phase and two-phase convective flow and heat transfer,

- Sub-cooled, saturated two-phase and vapor break flows over a range of break sizes,

- Structural heat transfer as a heat sink and as a heat source,

- Reactor kinetics and decay heat as a heat source,

- Cladding oxidation as a heat source.

The two-phase flow distribution includes the calculation of core void fraction, collapsed liquid level, and two-phase mixture levels in the core and balance of the RCS, where appropriate. A full boiling curve is used to describe the heat transfer in the core and SG, and elsewhere where appropriate. The heat transfer model considers single phase convection, nucleate boiling, critical heat flux (CHF) and post-CHF behavior. The appropriate flow regimes are considered, including single phase convection, two-phase co-current and counter-current flows, and flooding in the core, the RV, the RCS piping, the SG, and other primary and secondary system components.

3.4 Identify and Rank Key Phenomena and Processes

MHI described the process used to develop the US-APWR SBLOCA PIRT in Section 4.0 of Topical Report MUAP-07013-P. The PIRT developed for the US-APWR SBLOCA was based on the PIRT developed by Bajorek [TEC02] for a Westinghouse PWR. The ranking definitions for the phenomena were based on those developed by Boyack [TEC03].

In developing the PIRT, the phenomena were identified by major system components, as shown below in Table 1,"US-APWR SBLOCA PIRT High Ranked Processes and Phenomena Summary", and a ranking was assigned for the respective periods of the SBLOCA using the definitions for "High, Medium, and Low" as defined by Boyack [TEC03]. Based on historical PWR experience, MHI assumed that the accident is a small cold leg break LOCA with a most limiting single failure associated with the safeguard system. When developing the PIRT it was

necessary to judge the relative importance of the different phenomena expected during the transient relative to a measure of merit and to assign a rank for each phenomenon during each of the event periods. For the SBLOCA PIRT, the main measure of merit that has been used in the past is the resulting PCT. MHI chose this measure of merit when evaluating and ranking the various phenomena of interest.

MHI divided the transient into several periods to identify various phenomena and provide importance rankings for them during the SBLOCA transient. Some phenomena that exhibit a significant importance in a certain period may not necessarily exhibit such significance in other periods. However, simulations of these significant phenomena are required to accurately predict the overall US-APWR transient response.

SBLOCA transients were divided into five phases: (1) BLD, (2) Natural Circulation, (3) Loop Seal Clearance, (4) Boiloff (BO) and (5) Core Recovery.

The length and existence of each phase depends on the break size and the performance of the ECCS. It was assumed that the break is a small break located at the bottom of the reactor cold leg. These phases are defined and described in Section 4.1.1, “Small Break LOCA Scenario,”, of this SE.

Once the high ranked phenomena were identified, MHI proceeded to establish a code assessment matrix that was designed to demonstrate that the M-RELAP5 code could adequately predict those phenomena during conditions representative of the periods where the phenomena were highly ranked.

The PIRT is summarized in the following table:

Table 1. US-APWR SBLOCA PIRT Important Processes and Phenomena Summary

Location	Process/Phenomena	High Rank[1]
A. Fuel rod	1. Stored Energy/Initial Stored Energy	
	2. Core Kinetics, Reactor Trip (fission power)	
	3. Decay Heat	X
	4. Oxidation of Cladding	
	5. Clad Deformation	
	6. Gap Conductance	
	7. Local Power	X
B. Core	8. Heat Transfer below the Mixture Level	
	9. CHF[2]/Dryout	X
	10. Uncovered Core Heat Transfer	X
	11. Rewet (Heat Transfer Recovery)	X
	12. Entrainment/De-entrainment	
	13. 3-D Flow	
	14. Mixture Level	X
	15. Flow Resistance	
	16. 3-D Power Distribution	X
	17. Top Nozzle/Tie Plate CCFL[3]	
C. Neutron reflector	18. Steam and Droplet Generation in Flow Holes	
	19. Water Storage and Boiling in Back Region	
	20. Heat Transfer between Back region and Core Barrel	
	21. Core Bypass Flow	

Location	Process/Phenomena	High Rank[1]
D. Upper head	22. Drainage to Core/Initial Fluid Temperature 23. Bypass Flow between Upper Head and Downcomer 24. Metal Heat Release	
E. Upper Plenum	25. Mixture Level 26.Drainage to Core 27. Entrainment/De-entrainment 28. Bypass Flow/Hot Leg-Downcomer Gap 29. Metal Heat Release	
F. Hot leg	30. Stratified Flow/Counter-flow 31. Entrainment/De-entrainment 32. Metal Heat Release	
G. Pressurizer and surge line	33. Mixture Level 34. Out-Surge by Depressurization 35. Metal Heat Release/Heater 36. Location/Proximity to Break	
H. Steam generator	37. Water Hold-Up in SG Inlet Plenum 38. Water Hold-Up in U-Tube Uphill Side 39. Primary Side Heat Transfer 40. Secondary Side Heat Transfer (Water Level) 41. Metal Heat Release 42. Multi-U-tube Behavior 43. Auxiliary Feedwater	X X X X
I. Crossover leg	44. Water Level in SG Outlet Piping 45. Loop Seal Formation and Clearing (Entrainment/Flow Regime/Interfacial Drag/Flow Resistance) 46. Metal Heat Release	X X
J. Reactor coolant pump	47. Coast-down Performance 48. Two-Phase Flow Performance 49. Reversal Flow of ECC Water 50. Metal Heat Release	
K. Cold leg	51. Stratified Flow 52. Condensation by Accumulator Water 53. Non-condensable Gas Effect 54. Metal Heat Release	
L. Accumulator	55. Large Flow Injection/Flow Resistance 56. Small Flow Injection/Flow Resistance 57. Interfacial Heat Transfer 58. Metal Heat Release 59. Injection of N_2 Gas Effect	
M. Downcomer/Lower Plenum	60. Mixture Level/Void Distribution 61. Metal Heat Release 62. ECCS Water/Mixing 63. 3-D Flow 64. DVI/SI Water/Flowrate 65. DVI/SI Water/Condensation 66. DVI/SI Water/Injection Temperature	X X

Location	Process/Phenomena	High Rank[1]
N. Break	67. Critical Flow	X
	68. Break Flow Enthalpy	X

[1]- “X” indicates process/phenomena ranked high for a least one of the five phases
[2] - CHF - critical heat flux
[3] - CCFL - counter-current flow limitation

3.5 Assessment Base

The M-RELAP5 assessment data base was described in Section 5.0 of Topical Report MUAP-07013-P. The following information regarding the tests, taken from public reports, were provided: the facility design, the scaling of the test to the US-APWR, range of the test conditions, the data to be compared, and the data uncertainty and known distortions.

The following seven tests were used for separate effects studies to validate and verify the models used in the M-RELAP5 code.

3.5.1 ROSA/LSTF Void Profile Test

ROSA/LSTF [DAT01] is a volumetrically-scaled (1:48) full-height model of a typical Westinghouse four-loop PWR. The facility includes a pressure vessel and two symmetric primary loops each one containing an active SG and an active coolant pump. The pressure vessel contains a 1104-rod (1008 electrically heated and 96 unheated rods), full-length (3.66m) bundle. The rod diameter and pitch are typical of a 17x17 fuel assembly. The heater rods are supported at ten different elevations by grid spacers with mixing vanes. The radial power distribution is uniform. The axial power profile is a chopped-cosine with a peaking factor of 1.495. The differential pressures are measured for the overall and seven of the vertical segments along the rod bundle. Approximately 500 thermocouples are installed in the bundle to measure fluid temperatures and rod surface temperatures. The maximum break size was designed to be 10 percent of the 1/48-scaled cold leg flow area of the referenced PWR.

3.5.2 ORNL/THTF Void Profile Test

Void profile tests were performed in the Thermal Hydraulic Test Facility (THTF) at Oak Ridge National Laboratory (ORNL) [DAT02]. The THTF is a large high-pressure non-nuclear thermal hydraulics loop. The system configuration was designed to produce a thermal-hydraulic environment similar to that expected in a SBLOCA. The THTF test section contains a 64-rod electrically heated bundle. The four unheated rods were designed to represent control-rod guide tubes in a nuclear fuel assembly. Rod diameter and pitch are typical of a 17x17 fuel assembly.

3.5.3 ORNL/THTF Uncovered Heat Transfer Test

Experiments of uncovered-bundle heat transfer test were performed at ORNL in the THTF [DAT02]. The test facility is described in Section 3.5.2 of this SE. The objective of the heat transfer testing was to acquire heat transfer coefficients and fluid conditions in a partially uncovered bundle.

3.5.4 ORNL/THTF High-Pressure Reflood Test

The experiments for the high-pressure reflood tests were performed in the THTF at ORNL [DAT03]. The test facility is described in Section 3.5.2 of this SE.

3.5.5 FLECHT-SEASET Forced-Reflood Test

A series of forced flow and gravity feed bundle reflooding tests and steam cooling tests were conducted on a heater rod bundle with dimensions typical of the 17x17 PWR fuel design [DAT04]. The purpose of these tests was to provide a reflooding database that could be used to develop or verify reflood prediction methods. These tests examined the effects of initial cladding temperature, variable stepped flooding rates, rod peak power, constant low flooding rates, coolant subcooling, and system pressure. The test section consisted of 161 heater rods and 16 thimbles. Sufficient instrumentation was installed such that mass and energy balances could be computed from the data. Data obtained in the experiments were rod cladding temperatures, turnaround and quench times, heat transfer coefficients, inlet flooding rates, overall mass balance, differential pressures and calculated void fractions in the test section, thimble wall and steam temperatures, and exhaust steam and liquid carryover rates.

3.5.6 UPTF Full-Scale SG Plenum CCFL Test

The Upper Plenum Test Facility (UPTF) [DAT05] simulates a four-loop German PWR, which is similar to a U.S. four-loop Westinghouse PWR. A full-size RV and piping (four hot legs and four cold legs) are included in UPTF. Emergency Core Cooling can be injected in the hot and/or cold legs of all four loops, or in the downcomer. One of the four loops contains break valves which are piped to a large containment simulator tank. The four SGs are simulated by four steam/water separators and the four RCPs are simulated by four passive, adjustable resistances. The RV upper plenum internals and top-of-core are full-scale replicas. The core is simulated by a steam/water injection system with 193 nozzles, one for each active fuel assembly that would be present in a PWR. UPTF was originally designed as an integral system test facility covering the end-of-blowdown, refill, and reflood phases of a LBLOCA. As discussed in the test report [DAT05], it has also proven very useful as a full-scale separate effects facility covering both LBLOCA and SBLOCA phenomena. UPTF can operate at up to 18 bar (260 psia) pressure and 220 ºC (428 ºF) temperature.

3.5.7 Dukler Air-Water Flooding Test

The facility [DAT06] consists of four major sections. The first section (the lowest section) is a 5-foot length of 2-inch inner-diameter (ID) plexiglass pipe where the upward flowing air is injected and stabilized prior to reaching the test section (also known as the third section). The second section is a 12-inch ID plexiglass pipe where air enters the test section from below and non-entrained water is collected and sent to the liquid measuring tank. The third section is a 13-foot high, 2-inch ID plexiglass pipe where the liquid water is injected and interacts with the upward flowing air. This third section is referred to as the test section. It contains four pressure and film thickness measuring stations and a liquid water entrance device. The fourth and final section is the exit section for removing the air, the entrained water and the liquid film upflow.

The following five tests were used for integrated effects studies to validate and verify the models used in the M-RELAP5 code:

3.5.8 ROSA-IV/LSTF Small Break (5 percent) LOCA Test (SB-CL-18)

The ROSA-IV/ LSTF [DAT07] is a 1/48 volumetrically-scaled model of a Westinghouse-type 3423 MWt four-loop PWR. It has the same major component elevations as the reference PWR to simulate the natural circulation phenomena, and large loop pipes (hot and cold legs of 207 mm (8.1 in) in diameter) to simulate the two-phase flow regimes and phenomena of significance in an actual plant. The equipment can be controlled in the same way as that of the reference PWR to simulate long term operational transients. It is designed to be operated at the same high pressures and temperatures as the reference PWR.

3.5.9 ROSA-IV/LSTF Small Break (10 percent) LOCA test (SB-CL-09)

A 10 percent cold leg break experiment (SB-CL-09) was conducted at the Large-Scale Test Facility (LSTF) on August 28, 1986 [DAT08]. The objective of the SB-CL-09 test was to clarify thermal-hydraulic phenomena especially for core cooling conditions under SBLOCA conditions with relatively large break sizes. A break orifice with an inner diameter of 31.9 mm (1.26 in) was used to simulate the 10 percent cold leg break. The high pressure injection system (HPIS) was assumed to fail.

3.5.10 ROSA/LSTF Small Break (17 percent) LOCA test (IB-CL-02)

A 17 percent cold leg break experiment (IB-CL-02) was conducted at the LSTF on September 10, 2009, at the MHI-JAEA joint program, simulating the 1.0 ft^2 cold leg break in the US-APWR.

The objective of the IB-CL-02 test was to confirm, experimentally, the thermal-hydraulic phenomena that are expected to occur in the US-APWR 1.0 ft^2 cold leg break [DAT09] scenario.

3.5.11 LOFT Small Break (2.5 percent) LOCA test (L3-1)

The LOFT integral test facility is described in Reference DAT10. The LOFT facility was scaled to represent a 1/60-scale model of a typical 1000-MWe (electric) commercial four-loop PWR. The unique feature of the facility was that the Reactor System had a UO_2 powered core. The entire nuclear core consisted of five square and four triangular fuel bundles with a total of 1300 fuel pins. The length of the core was 5.5 feet (1.68 m) instead of 12 feet (4 m), about one-half the length of typical reactor cores in commercial plants. However, this was the only compromise made in the nuclear fuel for the LOFT core. PWR 15x15 array fuel rod assemblies were used, complete with upper and lower end boxes and fuel rod spacer grids at five axial positions.

3.5.12 Semiscale Small Break (5 percent) LOCA test (S-LH-1)

Semiscale S-LH-1 test is described in Reference DAT11. The core power was maintained around 2000 kW and then decreased according to the preset decay heat curve following the reactor trip signal generated due to low pressurizer pressure. The specified initial pressurizer pressure was 15.47 MPa (2243.7 psia) and the intact and broken loops cold leg temperature was about 562 K (552 °F) and 564 K (556 °F), respectively, with 38 K (68 °F) of core ΔT. The nominal primary flow rates through the cold legs were 7.13 kg/s (15.72 lbm/s) and 2.35 kg/s (5.18 lbm/s) between intact and broken loops. The loop flow split was 3:1.

3.6 DEVELOPMENT OF THE RELAP5-3D-BASED FRAMEWORK FOR THE M-RELAP5 EM

3.6.1 Evaluation Model Structure

The EM structure includes the structure of the individual component calculational devices, as well as the structure that combines the devices into the overall EM. RG 1.203 describes the structure for an individual device or code, which consists of the following six parts:

(1) Systems and components: The EM structure should be able to analyze the behavior of all systems and components that play a role in the targeted application.

(2) Constituents and phases: The code structure should be able to analyze the behavior of all constituents and phases relevant to the targeted application.

(3) Field equations: Field equations are solved to determine the transport of the quantities of interest (usually mass, energy, and momentum).

(4) Closure relations: Closure relations are correlations and equations that help to model the terms in the field equations by providing code capability to model and scale particular processes.

(5) Numerics: Numerics provide code capability to perform efficient and reliable calculations.

(6) Additional features: These address code capability to model boundary conditions and control systems.

3.6.2 Selection of a RELAP5-Based Code with Inclusion of Appendix K Conservative Models

RELAP5 is based on a non-equilibrium, separated two-phase flow thermal hydraulic approach with additional models to describe the behavior of the components of reactor systems including heat conduction in the core and reactor coolant system, reactor kinetics, control systems and trips. The code also has generic and specialized component models such as pumps and valves. Special process models are included to represent those effects important in a thermal hydraulic system including form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and non-condensable gas transport.

MHI selected RELAP5-3D, a recent version of RELAP5, as the starting point for its development of M-RELAP5 for the US-APWR SBLOCA analyses to demonstrate compliance with the NRC rules and regulations.

RELAP5-3D includes several advanced user and modeling options. These options are not used in the US-APWR SBLOCA calculations or in the code-to-data comparisons. The most notable options are (1) the multi-dimensional thermal hydraulic component typically used to model the flow in the lower plenum, core, upper plenum and downcomer regions of an LWR and (2) the multi-dimensional neutron kinetics model. These options would be selected through user input.

M-RELAP5 uses a six-equation hydrodynamics model to describe the liquid, vapor, and non-condensable gases in the system. The six-equation model uses five independent state (thermodynamic fluid) variables with an additional equation for the non-condensable gas component. The state variables are (1) the pressure, (2) the vapor/gas void fraction, (3) the liquid-phase specific internal energy, (4) the vapor-phase specific internal energy, and (5) the total non-condensable mass fraction in the vapor/gas phase.

A hydrodynamic volume can contain liquid, vapor, or a mixture of the two. The vapor may also be a mixture of steam and non-condensable gases, and the liquid may contain dissolved boron. The liquid, vapor and non-condensable gases within a hydrodynamic volume are considered to be at the same pressure, but the liquid and vapor/gas mixture may have different temperatures. M-RELAP5 needs thermodynamic properties for single phase liquid, single phase vapor, and saturated states. The basic thermodynamic quantities needed are temperature, pressure, specific volume/density, internal energy, enthalpy, and entropy. Thermodynamic derivative quantities can either come directly from the equation of state for water or can be computed from properties taken from the equation of state.

M-RELAP5 uses the 1967 American Society of Mechanical Engineers (ASME) steam tables to calculate the basic properties for light water. When a non-condensable gas is present, M-RELAP5 uses a modified Gibbs-Dalton mixture for the vapor (real gas from the thermodynamic data) and an ideal non-condensable gas.

The M-RELAP5 thermal-hydraulic model solves eight field equations for eight primary dependent variables. The primary dependent variables are the pressure, the liquid and vapor phase-specific internal energies, the vapor/gas volume fraction (void fraction), the liquid and vapor phase velocities, the non-condensable quality, and the boron density.

An Eulerian boron tracking model is available in M-RELAP5 to simulate the transport of a dissolved component (solute) in the liquid phase (solvent). The solution is assumed to be sufficiently dilute so that there is a negligible impact on the liquid, and that one additional field equation for the conservation of the solute (i.e., boron) is included. It is not used for the US-APWR SBLOCA analyses.

Modeling of the reactivity feedback effects of boron or other soluble materials in the system is not within the approved capabilities of the M-RELAP5 model, and is a limitation on the use of M-RELAP5 for US-APWR SBLOCAs.

The constitutive relations in M-RELAP5 include models for defining flow regimes and flow regime related models for interphase friction, the coefficient of virtual mass, wall friction, wall heat transfer, interphase heat and mass transfer, and direct (sensible) heat transfer. Heat transfer regimes are defined and used for wall heat transfer. The models are summarized in Section 6.2.4, "Closure Relationships," in Topical Report MUAP-07013-P. The model details are described in the RELAP5-3D, "Code Structure, System Models, and Solution Methods" manual [TEC04]. The specific models and correlations used in the code are described in detail in the RELAP5-3D, "Models and Correlations" manual [TEC05].

MHI provided complete documentation for M-RELAP5 as follows:

(1) M-RELAP5 Code Supplementary Manual Volume 1: Code Structure, System Models and Solution Methods: MHI provided Reference [MHI26] to supplement the RELAP5-3D

"Code Structure, System Models and Solution Methods," [TEC04] which incorporates the technical descriptions of the MHI Appendix K changes to the computer program.

(2) M-RELAP5 Code Supplementary Manual Volume II: User's Guide and Input Requirements: MHI does not plan to issue this manual, because there are no changes from the original RELAP5-3D manual. Reference [MHI27] provides the description of the Appendix K data inputs.

(3) M-RELAP5 Code Supplementary Manual Volume III: Developmental Assessment Problems: MHI provided Supplementary Manual Volume III in Reference [MHI28].

(4) M-RELAP5 Code Supplementary Manual Volume IV: Models and Correlations: MHI provided Supplementary Manual Volume IV in Reference [MHI29].

(5) M-RELAP5 Code Supplementary Manual Volume V: User's Guidelines: MHI provided Supplementary Manual Volume V in Reference [MHI30].

3.7 Develop or Incorporate Closure Models

Section 7.0 in Topical Report MUAP-7013-P described the implementation of the required Appendix K models into the M-RELAP5 Code.

Some of the Appendix K requirements were satisfied by providing the appropriate input in the plant model. These included an appropriate plant nodalization together with the appropriate initial conditions, boundary conditions, and the selection of the proper code options. In some cases, sensitivity calculations were necessary to verify that these inputs would provide conservative analyses of the ECCS performance. However, some Appendix K requirements could only be obtained through the implementation of new models or the modification of existing RELAP5-3D models. Some of these models were also validated by the additional comparison with appropriate experimental data to confirm the acceptability of the models to SBLOCA EM calculations.

3.7.1 Appendix-K Compliant Models

The Appendix K requirements, and how MHI addressed each, are summarized in the following table:

Table 2. US-APWR Approach for Meeting Appendix K Requirements

Appendix K Requirement	Section	Acceptable Limits	MHI Approach for Acceptance
1. Steady state power level	I.A	Power level shall be at least 1.02 times the licensed power level.	Provide appropriate input.
2. Maximum peaking factor	I.A	Maximum peaking factor shall be that allowed by the technical specification.	Provide appropriate input.

Appendix K Requirement	Section	Acceptable Limits	MHI Approach for Acceptance
3. Power distribution shape	I.A	Power distribution shape and peaking factor - .combination giving highest PCT shall be considered.	Provide appropriate input.
4. Initial stored energy in fuel	I.A.1	Steady state temperature distribution and stored energy in the fuel shall be calculated for the burn-up that yield highest PCT.	Provide appropriate input. A gap conductance model consistent with the fuel design code added to M-RELAP5. **SE Section 4.6.1**
5. Fission heat	I.A.2	Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivity from temperature and voids shall-be given their minimum plausible values.	Provide appropriate input.
6. Actinide decay heat	I.A.3	The heat from actinide decay shall be calculated.	Provide appropriate input.
7. Fission Product decay heat	I.A.4	Fission product decay heat shall be 1.2 times the values for infinite operating time in the ANS standard 1971.	ANS standard 1971 was added to M-RELAP5. **SE Section 4.6.2**
8. Gamma energy redistribution	I.A.4	The fraction of the gamma energy deposited in the fuel shall be justified by a suitable calculation.	Provide appropriate input.
9. Metal water reaction rate	I.A.5	Influence of the metal/water reaction shall be calculated using the Baker-Just equation. The reaction shall be assumed not to be steam limited. The inside of the cladding shall be assumed to react after the rupture.	Baker-Just equation added to M-RELAP5. **SE Section 4.6.3**

Appendix K Requirement	Section	Acceptable Limits	MHI Approach for Acceptance
10. Reactor internal heat	I.A.6	Heat transfer from piping, vessel walls, and transfer non-fuel internal hardware shall be taken into account.	Provide appropriate input.
11. SG heat transfer	I.A.7	Heat transferred between primary and secondary systems through heat exchangers shall be taken into account.	Provide appropriate input.
12. Cladding swelling and rupture	I.B	Cladding swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.	Cladding swelling and rupture data for ZIRLO™ alloy added to M-RELAP5. **SE Section 4.6.4** Gap conductance calculation for rupture node added to M-RELAP5. **SE Section 4.6.4**
13. Break characteristics	I.C.1a	A spectrum of possible break shall be considered.	Perform sensitivity study.
14. Discharge model	I.C.1b	Two-phase discharge rate shall be calculated using the Moody model with at least three values of a discharge coefficient. Discharge coefficient will span 0.6 to 1.0 or even a lower value if a maximum PCT may be calculated at such values.	The Moody critical flow model was added to M-RELAP5. **SE Section 4.6.5** Perform sensitivity study.
15. ECC water bypass	I.C.1c	ECC water shall be subtracted from the reactor vessel inventory during the bypass period. The end-of-bypass definition shall be justified by suitable combination of analysis and experimental data.	Additional assessment provided. **SE Section 4.6.8**

Appendix K Requirement	Section	Acceptable Limits	MHI Approach for Acceptance
16. Noding near break and ECC water injection points	I.C.1d	Noding near break and ECC water injection points shall be chosen to permit a reliable analysis of the thermodynamic history in these regions.	Provide appropriate input. Perform sensitivity study.
17. Frictional pressure drop	I.C.2	The frictional losses shall be calculated using models that include Reynolds number dependency, and realistic two-phase friction multipliers that have been adequately verified.	Provide appropriate input.
18. Momentum equation	I.C.3	Momentum equation shall include temporal change of momentum; momentum convection; area change of momentum flux; momentum change due to compressibility; pressure losses due to wall friction, and area change; and gravitational acceleration.	Provide appropriate input.
19. Critical heat flux	I.C.4	Correlations developed from appropriate steady state and transient-state experimental data are acceptable. The computer programs shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations.	CHF correlation incorporated in RELAP5-3D satisfied this requirement, with modifications in M-RELAP5. **SE Section 4.6.6** Additional validation was performed.
20. Return to nucleate boiling	I.C.4e	After CHF is predicted during blowdown, the calculation shall not use nucleate boiling heat transfer correlations subsequently during the blowdown.	Logic to prevent return to nucleate boiling during blowdown added to M-RELAP5. **SE Section 4.6.6**

Appendix K Requirement	Section	Acceptable Limits	MHI Approach for Acceptance
21. Post-CHF heat transfer	I.C.5	Transition and film boiling correlation, compared to applicable steady-state and transient-state data, shall be shown to predict values of heat transfer coefficient equal to or less than the mean value of data throughout the range of parameters for which the correlations are to be used. The Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result will no longer be acceptable.	Post-CHF heat transfer correlation incorporated in RELAP5-3D satisfied this requirement. Additional validation was performed. **SE Section 4.6.6**
22. Return to transition boiling	I.C.5b	Transition boiling heat transfer shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300 ºR.	Logic to prevent return to transition boiling during blowdown added to M-RELAP5. **SE Section 4.6.6**
23. Pump modeling	I.C.6	The pump model for the two-phase region shall be verified by applicable two-phase pump performance data.	Provide appropriate input.
24. Core flow distribution	I.C.7	The flow rate through the hot region of the core during blowdown shall be calculated as a function of time considering cross flow between regions and any flow blockage due to cladding swelling or rupture.	Provide appropriate input.
25. Single failure criterion	I.D.1	The most damaging single failure of ECCS equipment shall be considered.	Perform sensitivity study.

Appendix K Requirement	Section	Acceptable Limits	MHI Approach for Acceptance
26. Containment pressure	I.D.2	The containment pressure used during reflood shall not exceed a pressure calculated conservatively for this purpose.	Atmospheric pressure is applied as a boundary condition for the containment back pressure.
27. Reflood rate	I.D.3	The rate of reflooding of core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of reactor systems.	Advanced accumulator model added to M-RELAP5. **SE Section 4.6.7**
28. ECC water/steam interaction	I.D.4	The thermal-hydraulic interaction between steam and all ECC water shall be taken into account in calculating the core reflooding rate.	Provide appropriate input.
29. Refill/Reflood heat transfer	I.D.6	For reflooding rates of 1 in/s or higher, heat transfer shall be used based on applicable experimental data. When reflooding rates are 1 in/s or less, heat transfer calculation shall be based on the assumption that cooling is only by steam.	Reflood rate< 1 in/sec **SE Section 4.6.9** Reflood rate ≥ 1 in/sec Additional validation was performed. **SE Section 4.1.4.4**

* Revision 1 - Requirements 27 and 29 required additional validation for the reflood model

3.7.2 Advanced Accumulator Model

Appendix K item I.D.3 requires the reflood rate to be calculated by an acceptable model that takes into consideration the thermal hydraulic characteristics of the core and the RCS.

An advanced accumulator design is used in the US-APWR [MHI25]. The unique feature of the advanced accumulator design is the ability to control the injection flow rate using a flow damper. The advanced accumulator is designed to initially inject a large amount of coolant just after activation that compensates for the loss of coolant from the LOCA. After the initial high flow period, the advanced accumulator will inject water at a small flow rate for longer-term cooling.

MHI added a model for the advanced accumulator into M-RELAP5. The total resistance coefficient, or pressure loss from the accumulator exit to the RCS, is determined from the accumulator flow rate coefficient and the resistance coefficient from the injection piping. The

accumulator flow rate coefficient is a function of a cavitation factor and the water level in the accumulator. The accumulator flow rate coefficient is calculated from empirical correlations obtained from test data, which covered the range of applicability for the US-APWR design. The empirical correlations for the accumulator flow rate coefficients were derived separately for the large and the small flow rate injections as a function of the cavitation factor. The advanced accumulator model, as coded, was described in Appendix D in Topical Report MUAP-07013-P.

3.8 Assessment and Validation of the M-RELAP5 EM

Section 8.0 in Topical Report MUAP-07013-P provided the MHI assessment of M-RELAP5 for the phenomena that were ranked “high” in the PIRT. Section 8.1 provided the results of the M-RELAP5 comparisons with the Separate Effects Tests (SETs). Section 8.2 provided the results of the M-RELAP5 comparisons with the Integral Effects Tests (IETs). Section 8.3 provided a summary of the capability of the field equations to represent the processes and phenomena in the PIRT and the ability of numeric solution to approximate the equation set. Section 8.4 provided an assessment of the general application of the code to the US-APWR for SBLOCA conditions including an assessment of the different systems and components, constituents and phases, field equations, and numerics.

The phenomena that were ranked “high” in the PIRT were evaluated by comparisons to test data using the M-RELAP5 (Version 1.6) code.

The following phenomena were evaluated: CHF/core dryout, uncovered core heat transfer, rewet, core mixture level, water hold up in SG primary side, SG primary and secondary heat transfer, water level in the SG outlet piping, loop seal formation and clearance, downcomer mixture level/downcomer void distribution.

The following seven SETs and five IETs were analyzed with M-RELAP5:

SETs

- ROSA/LSTF Void Profile test
- ORNL/THTF Void Profile test
- ORNL/THTF Uncovered heat transfer test
- ORNL/THTF Reflood test
- FLECHT-SEASET Reflood test
- UPTF SG plenum CCFL test
- Dukler Air-Water Flooding test

IETs:

- ROSA/LSTF small break (5 percent) LOCA test (SB-CL-18)
- ROSA/LSTF small break (10 percent) LOCA test (SB-CL-09)
- ROSA/LSTF small break (17 percent) LOCA test (IB-CL-02)
- LOFT small break (2.5 percent) LOCA test (L3-1)
- Semiscale small break (5 percent) LOCA test (S-LH-1)

3.9 Scaling Report

The EM Development and Assessment Process, Step 6 of RG 1.203, requires a scaling analysis to "…demonstrate the relevancy and sufficiency of the collective experimental database for representing the behavior expected during the postulated transient …." Since MHI stated that it followed RG 1.203, the NRC requested MHI to perform a scaling analysis in RAI 1-2 [NRC12]. MHI responded [MHI05] and produced a scaling report in three parts [MHI13, MHI14, MHI15]. These reports were reviewed by the staff. The staff generated RAIs regarding the scaling evaluation provided by MHI [NRC09]. MHI provided responses [MHI16] and a revised scaling analysis [MHI17], which incorporated these RAIs.

The quantitative scaling analyses were based on the hierarchical two-tiered scaling (H2TS) methodology [TEC06]. The IET and SET facilities and experimental data were evaluated using the top-down and bottom-up approaches to determine whether similar thermal-hydraulic behaviors expected in the US-APWR were also observed in the scaled test facilities. The top-down scaling approach evaluated the global system behaviors and system interactions from the IETs, and addressed the similarity between the IETs and the US-APWR. The bottom-up scaling analyses addressed the issues identified in the US-APWR SBLOCA PIRT related to localized behaviors, where the SETs were examined.

Due to the similarity of the US-APWR design to the four-loop Westinghouse design, the top-down scaling analysis was limited to a confirmatory approach. The top-down scaling exercise demonstrated that, for each phase in the SBLOCA event, the relative rankings of non-dimensional coefficients between the plant and the test facility were preserved. This showed that the plant has no new or different system interactions from those exhibited in the test facilities.

The scope of the top-down scaling study was limited to the SBLOCA scenarios resulting in the highest PCTs. The top-down scaling analysis showed that the same dominant processes and phenomena occur in the US-APWR and the IETs; thus, the experimental data are acceptable for validation of the M-RELAP5 code for SBLOCA analysis.

When any scaling distortion was identified, due to differences in the non-dimensional coefficients, configuration and/or initial/boundary conditions between the IET and US-APWR, the effects of the distortion were evaluated. The scalability of locally important processes and phenomena that were not readily identifiable as part of the global behavior in the top-down scaling were examined with bottom-up scaling analyses, which included consideration of the experimental data for the SETs.

The M-RELAP5 code scale-up capability was examined using the bottom-up and top-down approaches to assess the adequacy of the US-APWR SBLOCA EM. The scalability of the models and/or correlations specific to the locally important processes and phenomena were evaluated based on the applicable range of the SET database. This scalability evaluation was limited to whether the specific model or correlation was appropriate for application to the configuration and conditions for US-APWR SBLOCAs. The scalability of the integrated M-RELAP5 code predictability, both for the US-APWR SBLOCAs and the IETs, was assessed using the top-down approach. This evaluation was performed to determine if the code calculations for the US-APWR SBLOCA and the IETs exhibit unexplainable differences that may indicate experimental or code scaling distortions.

3.9.1 Top-down Scaling

The top-down approach considered the system as a whole. Since no active part of the system was excluded, the top-down scaling provided a comprehensive understanding of the integral system response occurring during the accident scenario. The top-down scaling approach in the H2TS methodology proceeded from the whole system (reactor and/or plant) to the system components (reactor core, pressurizer, SG, RCP, ECCS, piping, etc.), to the constituents (fluid), to the phases (liquid and vapor), and to the fluid fields (continuous and dispersed flow regimes). One scaling group was generated for each transfer process between media at every level in the system's hierarchy.

Prior to the quantitative evaluation, the method identified the system to be addressed, and divided the transient and accident progression into several phases. The system response of interest in each phase was represented by the governing conservation equations, which account for the primary phenomena with a few simplified equations and lumped volume(s). The equations were mathematically non-dimensionalized and the non-dimensional groups, a set of non-dimensionalized coefficients characterizing the system response, were defined. The data from the plant and from the experimental test facilities were then used to evaluate the non-dimensional groups. The non-dimensional groups for the plant and test facility were compared to each other to quantitatively evaluate the applicability of the test data to the plant behavior. The magnitudes of the numerical values of the non-dimensional groups were used to determine the relative importance of the associated mechanisms. The applicability evaluation was based on the same mechanisms being dominant in the test facility and the plant, and on the relative ranking of the other important mechanisms.

3.9.2 Bottom-up Scaling

The bottom-up scaling approach was used to evaluate the similarity of the processes and phenomena of interest between the test facilities and the plant. This scaling approach assesses the applicability of the models and correlations implemented into the code; that is the bottom-up scaling was used at the local and/or component levels, not for the system level.

One of the methods used for the scaling study was the power-to-volume scaling, where the most important consideration was to preserve power and flow distribution as well as the time scale of thermal-hydraulic behaviors. Each component of the system was evaluated by determining the fluid volume ratio between the test facility and the plant, and agreement of the volume ratio with the facility-to-plant power ratio provided good scalability from the perspectives of time scale, fluid mass and energy distributions, velocities, acceleration, and length.

In the application of the power-to-volume scaling, it was necessary to consider several scaling effects and inherent deficiencies of the scaling criterion. In practice, it is generally impossible to simultaneously preserve length, elevation, area, volume, and pressure drop between the test facility and plant. For example, even if the test facility piping is well scaled based on the power-to-volume ratio concurrently with the full length and elevation, the hydraulic diameter differs from the actual plant, resulting in the different hydraulic resistance, and possibly in different flow regime characteristics. Therefore, in some cases scaling techniques based on the non-dimensional parameters representing flow characteristics were applied in the bottom-up approach.

4.0 Technical Evaluation

4.1 PIRT and Validation Plan

4.1.1 Small Break Loss-Of-Coolant Accident Scenario

MHI divided the transient into several periods to identify various phenomena and provide importance rankings for them during the SBLOCA transient. Some phenomena that exhibit a significant importance in a certain period may not necessarily exhibit such significance in other periods. However, simulations of these significant phenomena are required to adequately predict the overall US-APWR transient response.

SBLOCA transients were divided into five time periods: (1) Blowdown, (2) Natural Circulation, (3) Loop Seal Clearance, (4) Boiloff and (5) Core Recovery. The length of each time period depends on the break size and the performance of the ECCS. It was assumed that the break is a small break located at the bottom of the reactor cold leg.

(1) Blowdown

Following initiation of the break, the RCS primary side rapidly depressurizes until flashing of the hot coolant into steam occurs. Reactor trip is initiated on the "pressurizer low-pressure" set point of 1860 psia (12.82 MPa). Closure of the condenser steam dump valves isolates the SG secondary side and the SG secondary side pressure rises to the safety valve set point, and the steam is released through the safety valves. A SI signal is generated at the time that the pressurizer pressure decreases to the "pressurizer low-low pressure" set point at 1760 psia (12.13 MPa), and the SI initiates after a set delay time.

The coolant in the RCS remains in the liquid phase throughout most of the blowdown period, and towards the end of the period, steam begins to form in the upper head, upper plenum and hot legs. The rapid depressurization ends when the pressure falls to just above the saturation pressure of the SG secondary side at the safety valve set point. At that time, the steam generation rate in the upper regions of the core and in the upper plenum increases. The break flow is single-phase liquid phase throughout the blowdown period.

(2) Natural Circulation

When the blowdown period ends, the RCS pressure settles slightly above the SG secondary side pressure. Two-phase natural circulation is established through the RCS loops as the decay heat is removed by heat transfer (via condensation and convection) to the SG secondary side. The pressure rise in the secondary side is suppressed by steam venting through the secondary side safety valves. Auxiliary feedwater flow is initiated to maintain the secondary side liquid inventory. As more coolant is lost from the RCS through the break, the loop flow velocity decreases and natural circulation is broken. Steam accumulates in the downhill side of the SG tubes and the crossover leg and natural circulation flow stops when the pump suction piping (loop seals) fills with a phase liquid plug.

(3) Loop Seal Clearance

The loop-seal-clearance period starts when natural circulation ends. The period ends when the liquid level on the downhill side of the steam generator reaches the elevation of the loop seal and steam is vented towards the break.

With the loop seals present, the break remains covered with water. The RCS coolant inventory continues to decrease and steam volume in the RCS increases. During loop seal formation, the hydrostatic pressure difference that develops in the SG tubes depresses the liquid level in the core. This phenomenon is due to the difference in void fraction and mixture densities on the two sides of the SG. The uphill side of the SG is in countercurrent flow, with steam flowing upwards and liquid flowing downwards. The downhill side experiences co-current flow, with both phases flowing downwards. The mixture density is higher in the uphill side compared to the downhill side, which may generate a considerable hydrostatic pressure difference due to the height of the tubes.

This pressure difference is transmitted to the two-phase level in the core through the hot leg. As a result, the core is pressurized relative to the downcomer and a considerable portion of core inventory may be forced out from the core. If, during this process, the core mixture-level drops below the top of the core, a core uncovery occurs, and the cladding temperature in the upper part of the core begins to heat up. The core uncovery can be rapid and deep, but is short in duration. When the liquid level on the downhill side of the SG reaches the elevation of the loop seals, the seals clear and steam initially trapped in the hot portions of the RCS can be vented to the break.

The break flow changes from initially a low-quality mixture to primarily steam. As the pressure imbalances throughout the RCS are restored, the back pressure in the core is relieved. Then, the core liquid level is restored to the cold leg elevation with coolant flowing from the downcomer to the core.

(4) Boiloff

After the loop seal clears, the RCS primary side pressure falls below that of the secondary side due to the increase of the break flow quality, resulting in a lower mass flow rate but a higher volumetric break flow leaving the break. This changes the direction of heat transfer in the SG so that the secondary side begins to supply heat to the primary side. For a medium break size, the vessel mixture level may decrease as a result of the core boiling-off. This occurs because the RCS pressure is too high for the injection system to make up for the boiloff rate.

For the US-APWR, the flow from one SI pump is sufficient to match the boiloff rate for the case of a DVI line (3.4-inch (8.86 cm) inner diameter) guillotine break. Since two SI pumps are available, the SI is sufficient to maintain the vessel mixture level for a cold leg break of twice the area of the DVI line. For larger breaks, the core might uncover before the RCS depressurizes to the point where the SI pumps and accumulators deliver ECC water to the RCS at a higher rate than the break flow.

(5) Core Recovery

As the RCS pressure continues to fall, the SI flow increases and the accumulator eventually starts to inject such that total ECC flow exceeds the break flow. The vessel mass inventory increases and the core mixture level recovers. The transient terminates when the entire core is quenched and the ECC water delivery exceeds the break flow.

In its response to Request 4-1 [MHI03], MHI provided parameters to identify the boundaries of the five phases of the SBLOCA transient that indicate when one phase ends and the other begins.

Blowdown: The blowdown period starts from the initiation of the break. It ends when the primary system pressure has decreased to nearly equal to the secondary system pressure.

Natural Circulation: The natural-circulation period starts at the end of blowdown. It ends when the liquid flow rate at the top of SG U-tubes decreases to zero.

Loop Seal Clearance: The loop-seal clearance period starts when natural circulation ends. The period ends when the liquid level on the downhill side of the SG reaches the elevation of the loop seal and steam is vented towards the break.

Boiloff: The boiloff period starts after the loop seals clear and ends when the minimum RCS inventory is reached.

Core Recovery: The core recovery starts at the end of the boiloff period and ends when the fuel rod cladding in the entire core is quenched by a low-quality mixture.

The staff finds the definitions of the transient phases are acceptable for the development of the PIRT to identify the high ranked processes and phenomena that occur during a SBLOCA in the US-APWR.

4.1.2 Staff Evaluation of the PIRT

It appeared to the staff that stored heat in structures was neglected in each component because it was less than the decay heat. In its response to Request 4-4 [MHI03], MHI clarified the treatment of stored heat in structures.

Stored energy was not neglected; however, it was ranked as low importance. An analysis using M-RELAP5 was performed to calculate the integral of the RCS metal heat release to the fluid and the integral of the total decay power. Figure RAI-4-4.1, “Comparison between RCS metal heat release and decay power,” was provided to show the relative contributions for these heat sources. The integrated stored heat from RCS metal is small compared with the total heat transferred to the reactor coolant (less than 10 percent). The staff agrees with the ranking and importance of the stored heat.

The staff was unsure as to why stored energy in the fuel rods was ranked lower than decay heat since the temperature distribution in the fuel rod changes, releasing stored energy. In its response to Request 4-5 [MHI03], MHI explained why stored energy in the fuel rods was ranked lower than decay heat.

During a SBLOCA transient, the stored energy in the fuel rods is released directly to the coolant before the core uncovers and the fuel rods heat up. During the blowdown period, the heat-transfer coefficient between the fuel-cladding and the coolant is large, resulting in the stored energy in the fuel rods being quickly released, after the reactor trips and long before the PCT occurs. For this reason, stored energy in the fuel rods was ranked low. For the rest of the transient, the decay-heat coming out from the fuel will be dominant, so it was ranked medium importance during blowdown and natural-circulation periods. The decay-heat was ranked high during loop-seal clearance, boiloff and core-recovery periods, because of its effect on the

heatup rate when the core is uncovered. The stored energy itself was ranked low for the rest of SBLOCA transient. Therefore, the importance of stored energy was ranked lower in importance than the decay heat. The staff agrees with the ranking and importance of the stored energy in the fuel rods.

The staff was unsure why gap conductance was ranked low, especially when decay heat was rated high. In its response to Request 4-6 [MHI03], MHI explained why the gap conductance was ranked low.

The gap conductance governs the heat transfer of the stored energy in the fuel pellet to the cladding. As discussed in MHI's response to Request 4-5 [MHI03], stored energy was ranked low; as a result, the gap conductance was also ranked low. In a SBLOCA transient, core uncovery occurs gradually, and the fuel rod behavior at the uncovered location is quasi-static and the heat flux on the cladding surface from decay heat is relatively constant. Therefore, the cladding temperature during the uncovery period depends mainly on the vapor superheat and heat transfer between the vapor and the cladding, and not on the gap conductance. The staff agrees with the ranking and importance of the heat transfer of the stored energy in the fuel pellet.

The staff was unsure why pressurizer phenomena were not ranked in the PIRT. In its response to Request 4-14 [MHI05] MHI noted that it did not explicitly address pressurizer phenomena, but all pressurizer phenomena would be ranked low.

The staff was also unsure as to why the interfacial mass transfer, or flashing, was not identified as a phenomenon of interest since the pressurizer pressure is used as a parameter for the reactor trip and SI signal because vapor generation in the primary system has a strong influence on this pressure. In its response to Request 4-7 [MHI03], MHI agreed that flashing due to vapor generation in the pressurizer contributes to the pressurizer pressure lag and could influence when the reactor trip and SI set points were reached. The staff did not find the response to be adequate to address the concern because the effects of vapor generation in the pressurizer on the timing of safety signals could not be determined, and issued a follow-up request, Request 4-14 [NRC04].

In its response to Request 4-14 [MHI05], MHI performed a sensitivity calculation to quantify the effects of reduced flashing in the pressurizer since this would delay reactor trip and SI. The flashing was reduced by lowering the initial water temperature by 10°F (5.6°C) in the pressurizer volumes beneath the volume containing the water level. MHI provided comparisons of the cladding temperature, the pressurizer pressure, the core collapsed liquid level and the break mass flow rate for the US-APWR DCD base case (7.5-in break loop seal limiting PCT) and the sensitivity case with reduced flashing. The sensitivity analysis showed no significant impact on the PCT, with the sensitivity case showing a lower PCT (about 6°F (3.3C) lower). A minor effect was observed in the pressurizer pressure transient, in which the sensitivity case depressurizes slightly earlier during blowdown period. The transient profiles of core collapsed liquid level and break mass flow rate were almost identical. Based on this sensitivity analysis, the staff agrees that the ranking and importance of the vapor generation would be low. Furthermore, the sensitivity analysis showed the approach used by MHI for the SBLOCA evaluation provides a conservative PCT.

The MHI SBLOCA EM consists of a broken loop and a single, lumped loop representing the remaining three intact loops and the staff was concerned that the dynamics of individual loop seal clearing was not considered in the PIRT. In its response to Request 4-9 [MHI03], MHI

explained why the effect of loop dynamics (or asymmetric effects) was not included as one of the phenomena for consideration in the PIRT process since there was some uncertainty as to whether loop seal clearing will occur first in the broken loop or in the lumped intact loop (three loops combined).

The staff did not find the response to be adequate to address the concern because the EM could not determine if more than one loop seal could clear.

To address this concern, MHI performed a sensitivity study in which the three intact loops were individually modeled. This study, which is discussed in Section 4.2.1, "Loop Seal Modeling," of this SE, showed the lumped loop representation predicted a higher PCT, as compared to the four loop representation. Therefore, the staff finds that there is no need to add the treatment of the dynamics of individual loop seal clearing to the PIRT.

The staff would rank the large flow rate from the accumulator high during the recovery period, since this flow would have a direct impact on PCT. The applicant ranked the accumulator high flow rate medium during this period. In its response to Request 4-10 [MHI03], MHI explained why the large flow rate from the accumulator was ranked medium during the recovery period.

The large flow mode from the accumulator was ranked medium during the recovery period, which is the end of the SBLOCA transient, because only the accumulator provides additional ECC water injection since the high-head injection system has been started earlier during the BLD period. Calculated and measured results for ROSA-IV/LSTF Test SB-CL-18 showed that the majority of the core uncovery and heatup occurs before accumulator flow starts and that accumulator high flow rate was adequately modeled. Regardless of the ranking, the applicant modeled and assessed the accumulator high flow phenomenon. For this reason, the staff finds the ranking by the applicant acceptable.

The staff finds the US-APWR SBLOCA PIRT acceptable. The development of the PIRT was based on an acceptable approach [TEC02] and the ranking was based on an acceptable approach [TEC03]. The staff finds MHI has identified all the high-ranked processes and phenomena important to the US-APWR SBLOCA for the assessment of the M-RELAP5 code.

4.1.3 Validation of M-RELAP5 for US-APWR SBLOCA Analyses

The processes and phenomena that are modeled conservatively based on the Appendix K requirements are the following:

> Decay heat - Appendix K specified model is used with the conservative multiplication factor of 1.2.
>
> Local power, 3-D power distribution - Conservative input or assumption is applied (i.e., worst case peaking factors).
>
> Critical flow - Appendix K specified model is used. Verification was performed to determine that the implementation was correct. Break spectrum analysis was performed to determine the worst case break size and location.
>
> Break flow enthalpy - Sensitivity calculations were performed to determine the worst break orientation, top, bottom, and side connections. The break orientation influences

the two-phase flow behavior to be discharged from the break that eventually affects the energy to be discharged through the break.

DVI/SI water flow rate - Flow rate and temperature of the SI are treated by conservative input and assumptions to obtain limiting results.

The processes and phenomena associated with the core that were evaluated by comparisons to separate effects test data were the following:

- CHF/Dryout
  - ORNL/THTF Uncovered heat transfer test

- Uncovered Core Heat Transfer
  - ORNL/THTF Uncovered heat transfer test
  - ORNL/THTF High-Pressure Reflood Test
  - FLECHT-SEASET Forced Reflood Test

- Rewet (Heat Transfer Recovery)
  - ORNL/THTF High-Pressure Reflood Test
  - FLECHT-SEASET Forced Reflood Test

- Core Mixture Level
  - ROSA/LSTF Void Profile test
  - ORNL/THTF Void Profile test

The processes and phenomena associated with the SG primary side counter-current flow limitation (CCFL) model, which were evaluated by comparisons to separate effects test data, were the following:

- Water Hold-Up in SG Inlet Plenum
  - UPTF Full-Scale SG Plenum CCFL test (Kutateladze type correlation)

- Water Hold-Up in U-Tube Uphill Side
  - Dukler Air-Water Flooding test (Wallis type correlation)

The ROSA-IV/LSTF SBLOCA (5 percent) test SB-CL-18, ROSA-IV/LSTF SBLOCA (10 percent) test (SB-CL-09), ROSA/LSTF SBLOCA (17 percent) test (IB-CL-02), LOFT SBLOCA (2.5 percent) test (L3-1) and Semiscale SBLOCA (5 percent) test (S-LH-1) integral tests were used to evaluate the following processes and phenomena:

- Core CHF/Dryout

- Core Uncovered Heat Transfer

- Core Rewet (Heat Transfer Recovery)

- Core Mixture Level

- SG Water Hold-Up in SG Inlet Plenum

- SG Water Hold-Up in U-Tube Uphill Side

- SG Primary Heat Transfer

- SG Secondary Heat Transfer

- Crossover Leg Water Level in SG Outlet Piping

- Crossover Leg Loop Seal Formation and Clearing

- Downcomer/Lower Plenum Mixture Level/Void Distribution

The staff finds the validation plan developed by MHI to incorporate appropriate Appendix K models into the M-RELAP5 code and to assess M-RELAP5 against appropriate SETs and IETs acceptable because the validation plan addresses the high-ranked PIRT phenomena, covers all the major phases of SBLOCA, and includes the best or established tests based on comparison to previous SBLOCA validations.

4.1.4 Summary

The staff finds the development of the US-APWR PIRT followed acceptable practices. The US-APWR is similar to current operating large PWRs and no new processes or phenomena unique to the US-APWR were identified by MHI during its development. The staff agrees that there are no new processes or phenomena unique to the US-APWR.

The staff finds the PIRT rankings acceptable. The staff finds the validation plan developed by MHI to incorporate appropriate Appendix K models into the M-RELAP5 code acceptable. The staff finds the validation plan to assess M-RELAP5 against appropriate SETs and IETs acceptable because it addresses the high-ranked processes and phenomena.

4.2 Major Modeling Constructs

The US-APWR design includes both new features as well as improved components which will enhance the safety, operation and performance of the reactor system. The new design features and improved components include:

- DVI for SI Pumped SI flow

- Neutron Reflector (NR) to reduce the neutron damage to the reactor pressure vessel

- Refueling Water Storage Pit (RWSP) located in containment to eliminate sump switchover

- Model 100A RCP for increased reactor flow

- Advanced Accumulator for improved accumulator flow delivery

4.2.1 US-APWR Reactor Coolant Loop System Model

The RCS provides the reactor cooling and energy transport functions. The RCS consists of the RV, the SGs, the RCPs, the pressurizer, the reactor coolant pipes, and valves. The corresponding hydrodynamic nodalization is shown in Topical Report MUAP-07013-P, Figure 8.4-1, "Overall Nodalization of Primary System for US-APWR SBLOCA Analysis," and uses two loops to represent the RCS. One loop represents a single loop and includes the break location and the pressurizer. The other loop represents the remaining three intact loops.

Advanced Accumulator Modeling

In the advanced accumulator, injection flow rate is controlled by a variable resistance damper. The advanced accumulator is designed to provide initially a high injection flow rate, which compensates for the coolant lost in a LOCA event and allows refilling.

After the initial high flow rate period, the advanced accumulator provides longer term cooling at a lower flow rate after the vessel is refilled. The injection characteristics of the advanced accumulator have been determined by a full-height, one-half scale experimental facility. The injection characteristics of the advanced accumulator have been developed using correlations that relate a cavitation factor and a flow-rate coefficient. The existing accumulator model in RELAP5-3D could not simulate these injection flow rate characteristics. Therefore, MHI incorporated an advanced accumulator model into M-RELAP5 for the US-APWR, as described in Section 4.6.7 of this SE.

Safety Injection and DVI Injection Modeling

The DVI SI system design includes four trains to inject coolant directly into the RV. To simulate the DVI performance, it was necessary to model the initiation of injection by an SI signal, the injection characteristics of an SI pump, the enthalpy of injected coolant, and the location of injection. M-RELAP5 provides flexible modeling functions allowing the DVI to be simulated. The modeling scheme is equivalent to that for existing PWR designs using a cold-leg injection, except for the location of injection.

Steam Generator Modeling

The modeled regions of the SG primary and secondary sides were illustrated in Topical Report MUAP-07013-P, Figure 8.4.2-3, "Modeling Regions of Steam Generator." The primary side of the SG consists of an inlet plenum, tubes and an outlet plenum.

In its response to Request 8-5 [MHI06], MHI addressed the modeling of the SG U-tubes to justify combining the different length SG U-tubes into a single flow path where differences in behavior between tubes of different lengths were not simulated. These differences could affect reflux cooling as well and the flow through the U-tubes, with some U-tubes reversing flow.

The non-uniform behavior of SG U-tubes under steady-state, natural circulation conditions during the post-LOCA period has been investigated by the Japanese Atomic Energy Research Institute (JAERI) [TEC07]. The non-uniform behavior observed in the ROSA/LSTF experiments included: (1) reverse flow in some U-tubes, (2) cyclic fill and dump, and (3) stagnant vertical stratification. Code calculations with the single U-tube model might not be able to capture these non-uniform phenomena, resulting in an underestimated prediction of the heat transfer from the primary system to secondary system. The JAERI reference reports that RELAP5/MOD3 calculations with multiple U-tube models were able to capture the non-uniform phenomena.

MHI expected more uniform behavior during the natural circulation period in US-APWR SBLOCAs because a relatively stronger driving force works on the coolant flow through the SG U-tubes due to the higher core decay heat during the post-LOCA natural circulation period. MHI conducted a sensitivity calculation based on multiple U-tube noding. For the multiple U-tube noding, three paths with different flow lengths were modeled. The flow area in each path was one-third of the total flow area in the base model.

Comparisons between the single and multiple U-tube models for the 7.5-in cold-leg break case showed the two-phase natural circulation from the hot-leg to cold-leg continues longer in the longest U-tube channel, particularly in the broken loop, even when a loop seal formed in the crossover leg.

The heat transfer area to the SG secondary side was larger in the longest U-tube channel than in the other channels, and relatively larger steam condensation occurred there. This caused the longer two-phase natural circulation in the sensitivity calculation, particularly in the broken loop, which lessened the core level depression during the loop seal period. Consequently, no significant fuel cladding heatup occurred when the multiple U-tube model was applied. The sensitivity results indicated that the single U-tube model conservatively predicts the core level and the resultant PCT during the loop seal period.

Results for the 1.0 ft^2 cold-leg break case showed less sensitivity as compared with the results for the 7.5 in break case. In the 1.0 ft^2 break case, the timing of the CCFL break down was more significant than the SG condensation, and earlier CCFL break down was observed in the longest U-tube channel in the intact loop. This resulted in a slightly faster depressurization in the RCS and slightly higher flow rates from the advanced accumulator when the multiple U-tube model was used. Although the sensitivity was small, the comparison indicated that the single U-tube model provides a conservative PCT result, 50ºF (27.8°C) higher in the single U-tube model.

MHI concluded the sensitivity calculation with the multiple U-tube model showed the adequacy of the single U-tube model for the US-APWR SBLOCA safety analyses. Based on the sensitivity studies, the staff finds the use of a single U-tube path conservative and acceptable for the M-RELAP5 US-APWR SBLOCA analyses.

Loop Seal Modeling

A loop seal forms when the two-phase natural circulation loop flow is not sufficient to carry the steam down through the pump suction piping. Steam begins to collect on the downhill side of the SG reducing the gravitational head needed to maintain the liquid level in the core. Core uncovery may occur during this period. When the water level on the downhill side of the SG is depressed to the seal elevation, the seal clears and the core is recovered.

In its response to Request 8-4 [MHI06], MHI addressed the concern that it is highly unlikely that all three intact loop seals will clear simultaneously. With the three intact loops combined, the MHI methodology cannot predict clearing of two or three loops. When a second loop seal clears, there may not be enough pressure difference to clear the third and/or fourth loop seals. There is also an uncertainty on how much liquid remains trapped in the loop seals following opening of a path for steam flow. The staff also noted that the assessment for loop seal clearing was based on only one test facility, ROSA-IV/LSTF.

To address the combined loop model, MHI performed sensitivity calculations using a four-loop RCS model with three separate intact loops to investigate the governing factor determining the number of loop seals clearing and to investigate the effects of the number of loop seals clearing and the uncertainty of the clearing time on the core thermal-hydraulics.

The investigation showed:

(1) The number of loop seals cleared is governed by the decay-heat level and the break flow rate, which is dependent on the break size. The loop seal PCT occurs for break sizes in the range from 3 to 10 inches and all 4 loops clear for break sizes greater than or equal to 6 inches. Uncertainty in the number of loop seals clearing was recognized for break sizes less than or equal to five inches.

(2) No heat-up was calculated for break sizes less than or equal to five inches even though there is uncertainty in the number of loop seals clearing. If the loop seals do not clear at the same time, the PCT becomes lower for the 6-inch and 7.5-inch break cases where the predicted heat-up occurs during the loop seal clearing period.

(3) MHI concluded that conservative results for the loop seal PCT were obtained from the combined nodalization of the three intact loops.

If the number of cleared loops is small, i.e., one or two, the amount of water from the crossover legs entering the core region decreases and heatup might be a concern. However, this is precluded in the US-APWR because the high pressure SI is designed with a high capacity and injected directly into the RV to prevent the heatup.

From the verification of M-RELAP5 using experimental data, the following observations were made:

(1) ROSA-IV/LSTF (5 percent break) data were well predicted by M-RELAP5. The data also indicated that the governing mechanism for the clearing loop number depended on both the decay-heat level and the break size.

(2) M-RELAP5 has a tendency to overestimate the residual amount of water in the crossover leg: that is, less water being supplied from the crossover leg into the core region, which results in a conservative prediction of core thermal-hydraulics during the core boiloff and core recovery phases.

To address the use of a single ROSA-IV/LSTF (5 percent equivalent cold leg break) test to assess loop seal clearing, MHI performed additional studies for ROSA-IV/LSTF tests with 0.5 percent [DAT12], 2.5 percent [DAT13] and 10 percent [DAT14] equivalent cold-leg breaks.

MHI compared the differential pressures along the crossover legs in the ROSA-IV/LSTF tests (0.5 percent, 2.5 percent, 5 percent and 10 percent cold-leg break). The differential pressure was coupled to the water accumulation in the crossover leg, and the number of clearing loops. The amount of residual water was dependent on the break size. All the loops are unlikely to be cleared for smaller break sizes and some amount of water will remain in the crossover leg. The amount of water accumulation is lower in the broken loop side for the 0.5 percent break case. This same trend was observed in the 2.5 percent break case. For the 5 percent and 10 percent break cases, all the loops were cleared. MHI concluded that M-RELAP5 can predict the number of loop seals cleared and the amount of residual water in the crossover leg.

MHI also compared the core differential pressure and the fuel cladding temperature for the four tests. M-RELAP5 predicted the core differential pressure reasonably well. M-RELAP5 conservatively predicted the fuel cladding temperature.

MHI also performed a sensitivity study for a UPTF test [DAT15]. The base nodalization corresponded to the US-APWR modeling approach. The sensitivity analysis used a finer nodalization for the crossover leg region. The test injection procedure was simulated in the analysis and the residual amount of water remaining in the crossover leg region was evaluated after terminating the vapor injection. M-RELAP5 reasonably predicted the qualitative relation between the residual amount of water and the steam flow rate, and overestimates the amount quantitatively.

MHI concluded this quantitative tendency means less water being supplied from the crossover leg into the core region, which corresponds to a conservative prediction of core thermal-hydraulics during the core boiloff and core recovery phases.

MHI concluded that the sensitivity calculations and additional comparisons to test data showed the adequacy of the two loop (broken loop and combined single loop for the three intact loops) model for the US-APWR SBLOCA safety analyses. Based on the sensitivity studies and the additional comparisons to test data, the staff finds the use of the two loop model conservative and acceptable for M-RELAP5 US-APWR SBLOCA analyses.

Refueling Water Storage Pit (RWSP) Modeling

The in-containment RWSP is the ECCS SI pump water source. The in-containment RWSP eliminates the need for the changeover from an injection mode to a recirculation mode for the SI system, and enhances the reliability of core cooling following a postulated accident. The RWSP may have an effect on a transient behavior during a SBLOCA event because of the increase in the enthalpy (temperature) of the coolant injected by the SI pumps. M-RELAP5 has the capability to simulate the enthalpy of the injected flow as a function of time or a function of integrated injection flow rate using a time-dependent volume similar to the simulation method used for the injection enthalpy in existing PWR designs.

Reactor Coolant Pump Modeling

The Model 100A RCP is the primary coolant pump for the US-APWR. The Model 100A RCP achieves high capacity and enhanced efficiency through a redesign of the impeller/diffuser configuration. M-RELAP5 incorporates the same pump model that has been developed for the RELAP3, RELAP4, RELAP5/MOD1 and RELAP5/MOD2. The pump characteristics in a transient are simulated by a homologous curve. The pump coast-down is calculated using the angular momentum equation with the torque and the momentum of inertia as input data for the calculations. The flow resistance after the shutdown can be simulated through the input because it is determined by the characteristics of homologous curve corresponding to the condition during shutdown. Therefore, the M-RELAP5 code has the capability to simulate the Model 100A RCP.

4.2.2 US-APWR Reactor Pressure Vessel and Internals Model

The vessel is modeled using a combination of standard RELAP5 thermal-hydraulic components. The downcomer, core bypass, and core are described using a combination of annulus and pipe components.

The major flow paths, i.e. through the flow holes in the lower core support, and leakage paths are modeled by the combination of hydrodynamic components and volumes shown in the vessel nodalization diagrams. The flow areas and flow resistances for each of these flow paths are described through input and represent the actual geometries and characteristics of each structure. The flow characteristics within each of the flow paths can also be represented by selecting the appropriate modeling options. The selection of the appropriate modeling options for the US-APWR analysis were based on the RELAP5-3D recommended user guidelines as defined in references [TEC08 and TEC09]. The guidelines appropriate for the Westinghouse design of PWRs were used.

Neutron Reflector

The neutron reflector is a stainless steel ring that replaces the baffle plate surrounding the reactor core in existing PWRs and is installed between the reactor core and a core barrel of the US-APWR. M-RELAP5 can model the neutron reflector structure's thermal response and its effects on the heat transfer to the reactor coolant. M-RELAP5 models the flow holes through the reflector as well as the coolant flowing in the holes.

Core Region

The thermal behavior of the fuel rods in the assemblies is described using standard RELAP5 heat structures. The core heat transfer is modeled with several heat structures in order to provide appropriate thermal responses to evaluate the core hydraulic behavior and the heat-up behavior in the high power rod. The radial nodalization within each fuel rod was based on the RELAP5-3D recommended user guidelines as defined in references [TEC08 and TEC09]. The guidelines appropriate for the Westinghouse design of PWRs were used.

The standard RELAP5 modeling options for fuel rods such as gap conductance and cladding deformation are not used for the M-RELAP5 calculations since these models have been replaced by the conservative Appendix K methods described in Section 4.3.3 of this SE.

Summary of the US-APWR Vessel and Internals Model

The RV was divided into the downcomer, the lower plenum, the core, the neutron reflector flow channel, the upper plenum, and the upper head as shown in Topical Report MUAP-07013-P, Figure 8.4.1-1, “Overall Nodalization of Primary System for US-APWR SBLOCA Analysis.”

An expanded view of the upper and lower portions of the RV nodalization is presented in Topical Report MUAP-07013-P, Figure 8.4.1-6, “Expanded View of Nodalization of Upper and Lower Plenum Regions of Reactor Vessel.” The lower plenum and upper plenum regions are described by a series of interconnected branch components. The modeling corresponds to the recommended guidelines for nodalization for a Westinghouse PWR presented in user guidelines for RELAP5-3D. The flow paths, flow areas, and flow resistances, however, correspond to the specific geometry of the US-APWR.

4.2.3 Summary of the US-APWR SBLOCA Model

The US-APWR reference input model has been developed following the RELAP5-3D general user guidelines for the modeling of PWRs. In particular, the model was developed following the approach recommended for PWRs similar to the Westinghouse design, since US-APWR has a configuration comparable to that of Westinghouse PWRs. The general components, special process models, and special models for heat structures that are used in the model are summarized in Topical Report MUAP-07013-P, Table 6.2.1-4, "General Hydrodynamic Components Applicable for US-APWR." In some cases, because of the relatively large number of volumes and junctions used in the input model, pipes and branches are used in combination with single volumes or single junctions for convenience in building the input model. Even though single volumes and junctions can always be used to define the hydrodynamic system since they are the basic components used in the solution of the balance equations, it is possible to simplify the input by using the corresponding pipes, branches and other components that have been developed for that purpose.

The nodalization of the US-APWR was developed following the general user guideline developed for RELAP5-3D. Studies for the break location and orientation were performed in Technical Report MUAP-07025-P [MHI24] and the cold leg bottom orientation was determined to be the limiting PCT location for the break. In addition, nodalization studies near the break and the DVI injection location were performed in Technical Report MUAP-07025-P. Nodalization studies for the SG U-tubes and crossover leg were also performed in Technical Report MUAP-07025-P. Additional nodalization studies were also performed, as discussed in Section 4.2.1, "Steam Generator Modeling," and "Loop Seal Modeling," of this SE. These nodalization studies were performed to comply with Appendix K requirements I.C.1a, I.C.1d, II.2 and II.3.

4.2.4 Break Pull-Through Model

One consequence of stratification in a large horizontal pipe is that the properties of the fluid flowing through a small flow path in the pipe wall (i.e., a small break), called an offtake, depends on the location of the stratified liquid level in the large pipe relative to the location of the flow path in the pipe wall. If the offtake is located at the bottom of the horizontal pipe, liquid will flow through the offtake until the liquid level starts to approach (but not reach) the bottom of the pipe, at which time some vapor/gas will be pulled through the liquid layer and the fluid quality in the offtake will increase. If the phase separation phenomenon is ignored, vapor/gas will be passed through the offtake regardless of the liquid level in the pipe. Likewise, if the offtake is located at the top of the pipe, vapor/gas will be flowing through the offtake until the liquid level rises high enough so that liquid can be entrained from the stratified surface. The flow quality in the offtake will decrease as the liquid level rises. If the phase separation phenomenon is ignored, liquid will pass through the offtake for all stratified liquid levels regardless of their height relative to the offtake. Lastly, if the offtake is located in the side of the large horizontal pipe, the same phenomenon of vapor/gas pullthrough or liquid entrainment will occur, depending on the elevation of the stratified liquid level in the pipe relative to the location of the offtake in the wall of the pipe.

The RELAP5-3D stratification entrainment/pullthrough model for horizontal volumes accounts for the phase separation phenomena and computes the mass and energy flowing through the offtake attached to a horizontal pipe when stratified conditions occur in the horizontal pipe. This model is sometimes referred to as the offtake model. This model is used in M-RELAP5 to address the break enthalpy requirement in Appendix K Section I.C.1b for SBLOCAs.

Correlations are included in M-RELAP5 for offtakes situated at the top, bottom, and side of the horizontal pipe. M-RELAP5, Version 1.4 did not permit use of the offtake model and the critical flow model at the same junction. Therefore, MHI introduced a phantom volume, referred to as a stub pipe, downstream of the break.

In DCD Tier 2, Section 15.6.5, Revision 1 [MHI31] the applicant reported on the results for three limiting SBLOCA cases:

- 7.5-inch upside break, the limiting break for PCT during the loop-seal clearance phase.

- 1-ft^2 upside break, the limiting break for PCT during the boiloff phase.

- 3.4-inch DVI line break, with only 1 train of SI system assumed to operate.

M-RELAP5, Version 1.4 was used to perform these analyses. The limiting case for PCT was the 1-ft^2 break at the top of the cold leg piping.

To confirm the M-RELAP5 results obtained by the applicant, the staff performed a series of audit calculations for the US-APWR SBLOCA [ISL02] using the RELAP5/MOD3.3 computer code [NRC13]. RELAP5/MOD3.3 is an advanced thermal/hydraulic simulation tool developed by the staff. Conservative assumptions were used in the RELAP5/MOD3.3 analyses similar to those used in the M-RELAP5 analyses. Decay heat was set at 120 percent of the ANS 1971 Standard. The single failure of one of the ECC trains was assumed.

In RAI CA-5 [NRC05], the staff noted that confirmatory runs with RELAP5/MOD3.3 showed a large difference in PCT (approximately 300°F (166.7°C) lower to 200°F (111.1°C) higher) depending on the geometry of the stub pipe (length and area). In its response to these RAIs [MHI08] and follow-up meetings [MHI32], MHI made two revisions to the M-RELAP5 code. Version 1.5 made modifications which allowed use of the off-take model and the critical flow model at the same junction, and therefore eliminating the need for the stub pipe. While use of the stub pipe was no longer necessary, MHI retained the stub pipe in the calculations performed with M-RELAP5, Version 1.5. The staff performed additional confirmatory calculations with the RELAP5/MOD3.3 computer code [ISL03]. These calculations again yielded significantly different PCT values compared to the M-RELAP5 results. In the process of investigating the reason for these differences, the staff obtained the M-RELAP5 source code and performed calculations with modified versions of that code and the RELAP5/MOD3.3 code. It was determined that the critical flow switching logic in the M-RELAP5 code was such that the required Moody critical flow model was not being used at all times when the break flow was two phases. Rather, the code was switching between the Henry-Fauske and Moody models. Therefore, the 10 CFR 50 Appendix K requirement to use the Moody critical flow model whenever the conditions at the break are two phase was not being met. MHI revised the switching logic and corrected several other minor code problems in a new version, M-RELAP5, Version 1.6. This version of M-RELAP5 was used to produce the results in Chapter 15.6.5 of the DCD.

The sensitivity cases in Technical Report MUAP-07025-P [MHI21] affected assessment cases in Topical Report MUAP-07013-P [MHI22] were also rerun with M-RELAP5, Version 1.6 to produce Revision 2 of each of these reports. MHI continued to use the stub pipe in the US-APWR plant calculations. However, with the critical flow-switching logic corrected, the variation of PCT with stub pipe geometry was significantly reduced from 250°F (139°C) to 38°F (21°C).

Also, with Version 1.6 the bottom of cold leg break became the limiting case rather than the top of cold leg break case, making the US-APWR results consistent with those of other PWRs where the bottom break is limiting. The calculations provided by the applicant for SBLOCA response in Revision 2 of DCD Section 15.6.5 [MHI] are now consistent with the confirmatory calculations and are acceptable.

4.3 Code Modifications

Modifications were made by MHI in the development of M-RELAP5 code, from the RELAP5-3D code, where necessary to comply with 10 CFR 50 Appendix K requirements.

4.3.1 Fuel Gap Conductance Model - Appendix K Section I.A.1

A revised fuel gap conductance model was implemented in M-RELAP5. The fuel gap conductance model was described in Section 7.1.2, "Gap Conductance Model," of Topical Report MUAP-07013-P. The model was implemented in M-RELAP5 to maintain consistency with the fuel design code. The staff evaluation of this model is provided in Section 4.6.1 of this SE.

4.3.2 Fission Product Decay Model - Appendix K Section I.A.4

The ANS 1971 standard for decay heat was implemented in M-RELAP5. The decay heat model was described in Section 7.1.3, "Fission Product Decay," of Topical Report MUAP-07013-P. The staff evaluation of this model is provided in Section 4.6.2 of this SE.

4.3.3 Metal Water Reaction Model - Appendix K Section I.A.5

The Baker-Just equation for the metal water reaction rate was implemented in M-RELAP-5. The Baker-Just equation was described in Section 7.1.4, "Metal Water Reaction Model," of Topical Report MUAP-07013-P. The staff evaluation of this model is provided in Section 4.6.3 of this SE.

4.3.4 Cladding Swelling and Rupture Model - Appendix K Section I.B

The cladding swelling and rupture model in RELAP5-3D was modified to incorporate the new cladding swelling and rupture model of the US-APWR cladding material, ZIRLO™, in M-RELAP5. A model to account for the effect of the cladding geometry change was implemented in M-RELAP5. The cladding swelling and rupture model was described in Section 7.1.5, "Cladding Swelling and Rupture Model," of Topical Report MUAP-07013-P. The staff evaluation of these models is provided in Section 4.6.4 of this SE.

4.3.5 Discharge Model - Appendix K Section I.C.1b

The Moody critical flow model was implemented in M-RELAP5. The Moody critical flow was described in Section 7.1.6, "Discharge Model," of Topical Report MUAP-07013-P. The staff evaluation of the model is provided in Section 4.6.5 of this SE.

4.3.6 Critical Heat Flux (CHF) Model - Appendix K Section I.C.4

The CHF model in RELAP5-3D was modified in M-RELAP5 to account for low flow, high void fraction conditions based on observation from studies performed by MHI on the ROSA-IV/LSTF. With the current RELAP5-3D model, CHF was not predicted during the loop seal clearance period. In addition MHI implemented a modified bundle factor logic used prior to reactor scram. With the current RELAP5-3D model, CHF was predicted to occur earlier than expected. The CHF model was described in Section 7.1.7, “Critical Heat Flux and Post-CHF Heat Transfer Model,” in Topical Report MUAP-07013-P. The staff evaluation of this model is provided in Section 4.6.6 of this SE.

4.3.7 Prevent Return to Nucleate Boiling Model - Appendix K Section I.C.4e

A heat transfer mode selection logic to prevent the return to nucleate boiling once CHF has been predicted during blowdown was implemented in M-RELAP5. The model logic was described in Section 7.1.7.6, “Prevent Return to Nucleate Boiling and Transition Boiling,” in Topical Report MUAP-07013-P. The staff evaluation of this model is provided in Section 4.6.6, “Prevent Return to Nucleate Boiling and Transition Boiling,” of this SE.

4.3.8 Prevent Return to Transition Boiling Model - Appendix K Section I.C.5b

A heat transfer mode selection logic to prevent the return to transition boiling after the cladding surface superheat exceeds 300 ºF during blowdown was implemented in M-RELAP5. The model logic was described in Section 7.1.7.6, “Prevent Return to Nucleate Boiling and Transition Boiling,” in Topical Report MUAP-07013-P. The staff evaluation for the model is provided in Section 4.6.6, “Prevent Return to Nucleate Boiling and Transition Boiling,” of this SE.

4.3.9 Advanced Accumulator Model - Appendix K Section I.D.3

The RELAP5-3D accumulator model was not adequate to model the characteristics of the US-APWR advanced accumulator. A new model was implemented in M-RELAP5 for the advanced accumulator to ensure the rate of core reflooding is calculated with an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of reactor systems. The advanced accumulator model was described in Section 7.2, “Advanced Accumulator,” in Topical Report MUAP-07013-P. The staff evaluation of this model is provided in Section 4.6.7 of this SE.

4.4 Validation

4.4.1 Separate Effects Tests

4.4.1.1 ROSA/LSTF Void Profile Tests

The ROSA/LSTF void profile tests provided data on the two-phase mixture level in the core, an important parameter for the evaluation of PCT through the periods of loop seal clearance, boiloff and recovery.

A series of experiments [DAT01] were performed at the ROSA-IV Large Scale Test Facility (LSTF) [DAT16 and DAT17] to measure the void fraction distribution in the simulated reactor core rod bundle under high-pressure low-flow conditions.

The ROSA-IV LSTF is a volumetrically-scaled (1:48) full-height model of a Westinghouse designed four-loop PWR. The staff finds the use of the ROSA-IV LSTF as a separate effects test for the assessment of the mixture level acceptable.

Eleven test cases for three different pressures were simulated:

- ST-VF-01A, ST-VF-01B, ST-VF-01C, ST-VF-01D
  1.0 MPa (145 psia)

- ST-NC-01, ST-NC-06E, SB-CL-16L
  7.3 MPa (1059 psia)

- ST-VF-01 E, ST-VF-01 F, ST-VF-01 G, ST-VF-01 H
  15.0 MPa (2176 psia)

The ROSA/LSTF void profile tests for the rod bundle region were simulated using M-RELAP5. The tests at 7.3 MPa (1059 psia) were selected for analysis with M-RELAP5 because the pressure during the loop seal and core uncovery periods is expected to be around this value, and the ability of M-RELAP5 to predict the void fraction profiles at this pressure is important. MHI concluded the calculation result for the 7.3 MPa (1059 psia) test cases show good agreement with the test data for both the axial void fraction profile and the averaged void fraction. Since the void fraction was reasonably calculated, the staff finds the mixture level model in M-RELAP5 acceptable for the high pressure conditions expected in US-APWR SBLOCAs when the PCT occurs during the loop seal clearing phase of the accident.

In its response to Request 8.1.1-2 [MHI04], MHI provided M-RELAP5 comparisons to test ST-VF-01D (1.0 MPa (145 psia)) to address the mixture level PIRT ranked of “high” during the boiloff and recovery periods, which would occur at lower pressures, particularly for a larger break size. This provided a comparison of the void profile prediction over the core height. In addition, a comparison of the averaged void profile for the four 1.0 MPa (145 psia) tests was provided.

MHI concluded the void fraction under the lower pressure condition was generally overestimated by M-RELAP5. An overestimation of the void fraction is conservative because it results in a higher calculated PCT. Since the void fraction was overestimated, the staff finds the mixture level model in M-RELAP5 acceptable for the low pressure conditions expected in US-APWR SBLOCAs when the PCT occurs during the boiloff and recovery phases of the accident.

In its response to Request 8.1.1-4 [MHI03], MHI addresses differences between the US-APWR SBLOCA core model and the model used to simulate ROSA/LSTF.

The M-RELAP5 code was used to simulate the core during the loop seal clearing, boiloff and core recovery period of the SBLOCA. During these periods, cross flow in the core is important because the core axial flowrate is small. Therefore, the distribution in the radial void is small and the spatial dependence of the void fraction distribution is small under these conditions. For tests ST-NC-06E and SB-CL-16L, the radial core power distribution was flat; therefore, it was

sufficient to model the core as a single channel. For test ST-NC-01, the radial core power distribution was not flat, but the effect on radial void fraction distribution was not important, as noted above.

The two-channel model was used in the US-APWR calculations because it would be non-conservative to use steam temperatures from the average channel in the hot channel when the core is uncovered. The two-channel model was not required for these ROSA/LSTF tests because the core was not uncovered. The staff finds the model used for the ROSA/LSTF tests, as a separate effects test, acceptable.

In its response to Request 8.1.1-5 [MHI03], MHI compared the US-APWR SBLOCA grid spacer model to the model used to simulate ROSA/LSTF.

The flow area of the bundle was applied at the grid spacer instead of the reduced flow area based on the modeling guidelines provided in Section 2.4.1 of Volume II of the RELAP5-3D manual [TEC08]. The form loss coefficient was adjusted to preserve the pressure drop across the grid spacer. Grid heat transfer enhancement effect was not considered. The same modeling of the grid spacer was applied to both the test and the US-APWR simulations.

In its response to RAI 8.1.1-7 [MHI07], MHI discussed the void fraction decrease at the highest node in the M-RELAP5 ROSA/LSTF ST-NC-06E prediction, since it was expected the void fraction would continuously increase with the core elevation.

MHI concluded the reason the void fraction was lower at the highest node of the core was due to the junction-based interface friction factor applied there. The highest node was connected to the upper plenum node, which has a larger flow area than the core-heated section. The change in the flow area caused a reduced mass flux, resulting in a smaller friction factor at the top of the core-heated section, and the smaller void fraction. A sensitivity calculation in which the flow area of the upper plenum node was changed to be identical to that of the core-heated section was performed by MHI. The void fraction continuously increased in the sensitivity case. It could not be confirmed whether the void fraction continuously increased or began to decrease near the top of the heated core, since no experimental data were available. However, this void fraction degradation appears only at the highest node of the core and there was no impact on the PCT. The noding scheme for the upper plenum was consistent between the experimental test calculation and the US-APWR SBLOCA calculation.

Since the void fraction was reasonably calculated for high pressure conditions and overestimated for low pressure conditions, the staff finds the mixture level model in M-RELAP5 acceptable for the US-APWR SBLOCA analyses.

4.4.1.2 ORNL/THTF Void Profile and Uncovered-Bundle Heat Transfer Tests

Prediction of two-phase mixture level or void fraction profile in the core is important during the loop seal, boiloff, and recovery periods of a SBLOCA. During these periods the two-phase mixture level can drop into the core. The fuel rod is covered by high void fraction, two-phase flow up to the two-phase mixture level. Above the two-phase mixture level, the core is essentially covered by a single-phase vapor, and the fuel rod heat transfer above the two-phase mixture level is less than that in the two-phase region. As a result, the cladding temperature increases rapidly above the two-phase mixture level. Good rod heat transfer is maintained below the two-phase mixture level. Therefore the prediction of the two-phase mixture level and

void fraction profile near and below the two-phase mixture level is important for the prediction of the PCT in a SBLOCA.

A series of small break experiments were conducted in the THTF at Oak Ridge National Laboratory (ORNL). These experiments included the two-phase mixture level swell tests and the uncovered-bundle heat transfer tests, which were performed under quasi-steady state conditions. The axial void fraction profile was obtained from differential pressure measurements in the two-phase mixture level swell test, and the fuel rod simulator (FRS) temperatures and vapor temperatures above the mixture level were measured in the uncovered-bundle heat transfer test. These tests were used to assess the M-RELAP5 code applicability to the prediction of SBLOCA mixture levels, void fraction distributions and fuel rod heat transfer.

The THTF is a large high-pressure non-nuclear thermal hydraulics loop. The system configuration was designed to produce a thermal-hydraulic environment similar to that expected in a SBLOCA. The scaling of the facility is full length with prototypical PWR dimensions. The staff finds the ORNL/THTF acceptable for use for the assessment of M-RELAP5 for the prediction of SBLOCA mixture levels, void fraction distributions and rod heat transfer.

The series 3.09.10 tests assessed with M-RELAP5 include the following:

Table 3. ORNL/THTF Test Data

	Pressure		Mass flux		Inlet temperature		Subcooling		Liner heat rate	
Test	MPa	psia	kg/s-m^2	lb/s-ft^2	K	ºF	K	ºF	kW/m	BTU/ft-s
J	4.20	609	12.93	2.65	480.3	404.9	46.1	83.0	1.07	0.309
K	4.01	582	2.22	0.45	466.5	380.0	57.2	103.0	0.32	0.092
M	6.96	1009	13.38	2.74	474.4	394.3	84.2	151.6	1.02	0.295
N	7.08	1027	4.33	0.89	473.1	391.9	86.7	156.1	0.47	0.136
AA	4.04	586	21.15	4.33	450.9	352.0	73.2	131.8	1.27	0.367
BB	3.86	560	9.44	1.93	458.2	365.1	63.2	113.8	0.64	0.185
CC	3.59	521	7.22	1.48	467.6	382.0	49.6	89.3	0.33	0.095
DD	8.09	1173	19.82	4.06	453.4	354.7	115.5	207.9	1.29	0.373
EE	7.71	1118	11.00	2.25	455.9	361.0	109.7	197.5	0.64	0.185
FF	7.53	1092	4.83	0.99	451.4	352.9	112.6	202.7	0.32	0.092

All of the tests were used to access the mixture level and void profile. Test J, K, M and N were used to access the fuel rod heat transfer above the two-phase mixture level.

The prediction of the void profile and the two-phase mixture level is important to predict the PCT in a SBLOCA. The M-RELAP5 code was assessed by the comparison with the ORNL/THTF two-phase mixture level swell test and the uncovered-bundle heat transfer test. MHI concluded the assessment showed that the M-RELAP5 code reasonably predicts these parameters. Based on a review of the plot comparisons provided by MHI, the staff agrees with this assessment.

The prediction of the rod heat transfer above the two-phase mixture level is also important to predict the PCT in a SBLOCA. The M-RELAP5 code was assessed by the comparison with the

ORNL/THTF uncovered-bundle heat transfer test. MHI concludes that the assessment showed that the M-RELAP5 code reasonably predicts the rod heat transfer above the two-phase mixture level. Based on a review of the plot comparisons provided by MHI, the staff agrees with this assessment.

Additionally, code verification results from low pressure test data obtained in the ROSA/LSTF ST-VF-01 (steady-state void profile at 1.0 MPa (145 psia)) and SB-CL-09 (10 percent cold leg break LOCA) [DAT14] tests, were referenced to address the concern, identified by the staff in Request 8.1.2-3 [NRC03], that the ROSA/LSTF and ORNL/THTF assessments provided in Topical Report MUAP-07013-P R0 did not cover the full range of pressure for the SBLOCA periods in which the core mixture level was ranked high. These integral effects tests provided comparison data and supported the conclusions regarding the M-RELAP5 code models for US-APWR SBLOCA mixture level, void fraction distribution and rod heat transfer for the low pressure conditions expected for the 1.0 ft^2 break. M-RELAP5 provided an acceptable prediction of these parameters.

In its response to Request 8.1.2-7 [MHI03], MHI discussed the implication of the observation that "in most cases the calculated void fractions are slightly larger than the experimental values," in relation to the mixture level, and whether this systematic deviation indicated a code deficiency.

M-RELAP5 determines the void profile based on the liquid-vapor interfacial shear derived by the Chexal-Lellouche drift-flux model [TEC10] for the rod bundle geometry. The calculated results showed that M-RELAP5 slightly overestimates the void fraction for the lower pressure conditions and reproduces the measurement for the higher pressure condition. An overestimation of the predicted void fraction generally tends to result in a higher two-phase mixture swell. However, its impact is limited, because the mixture-level is sensitive to the transition void fraction that defines the boundary from the churn flow regime (two-phase flow) to the mist flow regime (mixture-level generation). The transition void fraction is dependent on the pressure, coolant flow and void fraction, and affects the velocity slip between the liquid and the vapor. The liquid-vapor interfacial shear (slip) model for two-fluid flow is used in the M-RELAP5 code to determine where the mixture-level is formed.

MHI concluded that M-RELAP5 can predict the measured mixture-level, which shows the interfacial shear model in M-RELAP5 valid not only for the void profile prediction, but also for the transition void fraction to determine the mixture-level. The staff finds the void model in M-RELAP5 acceptable for US-APWR SBLOCA analyses.

In its response to Request 8.1.2-10 [MHI04], MHI explained the discrepancy between the data and M-RELAP5 predictions which showed the measured heat transfer coefficients in the vapor region generally increased rapidly with temperature increase while the calculated coefficients were generally unchanged.

In the experimental investigation, the total heat transfer was determined from the measured FRS thermocouples and steam vapor temperatures, and the radiative heat transfer (radiation to vapor) was calculated from an empirical method with the measured temperatures. The convective heat transfer component was derived by subtracting the radiative component from the total heat transfer data. The spatial variation in the M-RELAP5 heat transfer coefficient was influenced by two factors. First, the convective enhancement due to the abrupt flow area change at the grid position was not accounted for in the current M-RELAP5 modeling for the THTF analysis. This resulted in no rapid increase in the calculated heat transfer coefficient near

the grid position, which was observed in the measured data. Second, M-RELAP5 has no explicit radiative heat transfer model for the single-phase vapor convection mode. This significantly affects the spatial variation, because the radiative heat transfer increased with elevation in all the tests.

MHI provided comparisons of the test data to convective and radiative heat transfer components as part of its response. The comparisons showed that the measured convective heat transfer coefficient either increases or decreases with elevation, depending on the test, while the measured radiative component always increases with elevation. Furthermore, the spatial variation of the M-RELAP5 heat transfer coefficient tended to agree with the measured convective heat transfer coefficient, except for Test K. MHI suspects that uncertainty in the measured radiative heat transfer might distort the accuracy of the convective heat transfer, particularly in the upper portion of the test bundle in Test K, because the error in the radiative component was significantly larger than in the other tests. The comparisons also showed that the heat transfer coefficient computed by M-RELAP5 based on the Dittus-Boelter correlation was generally larger than the measured convective component.

However, by neglecting the radiative component, the M-RELAP5 heat transfer coefficient in the steam cooling region was generally less than the measured total heat transfer coefficient. MHI concluded this to be a conservative basis for the code application. The staff agrees that neglecting the radiative component is conservative for the US-APWR SBLOCA analyses and the use of the Dittus-Boelter correlation is conservative.

In its response to Request 8.1.2-10 [MHI04], MHI also addresses the concern that in three (J, M, N) of the four tests, the measured heat transfer coefficients showed sudden drops after peaking in the vapor region.

The THTF test report explained the sudden change in the heat transfer was due to the grid effect on the local heat transfer. This effect can be generally classified with the following mechanism: (1) convective enhancement, (2) grid rewet by droplet impingement, and (3) droplet breakup. In the THTF uncovered-bundle heat transfer test, since there were few droplets because of the low inlet coolant flow, the convective heat transfer enhancement from the abrupt flow area change at the grid position could be the dominant factor. This grid effect tends to be greater under higher Reynolds number conditions. This convective enhancement was not observed in the lower inlet flow test, Test K. In the M-RELAP5 modeling for the THTF test analysis, and for the US-APWR SBLOCA, the abrupt flow area change at the grid position was not considered and the code computes the vapor and liquid momentum sources at the numerical junctions (including the grid positions) to determine the void distribution.

Therefore, the convective enhancement due to the grid is not considered in the M-RELAP5 analysis. M-RELAP5 predicts a smaller heat transfer coefficient when compared to a case where any grid effect model is taken into account, leading to a higher peak cladding temperature value. MHI concluded this to be a conservative basis for the code application to safety analyses. The staff agrees that neglecting the convective heat transfer enhancement due to the grid is conservative and acceptable for the US-APWR SBLOCA analyses.

The prediction of the two-phase mixture level is important to predict the PCT in a SBLOCA. The M-RELAP5 code was assessed by comparisons with the ORNL/THTF two-phase mixture level swell tests and the uncovered-bundle heat transfer tests. The prediction of the rod heat transfer above the two-phase mixture level is also important to predict the PCT in a SBLOCA. The M-

RELAP5 code was assessed by the comparisons with the ORNL/THTF uncovered-bundle heat transfer test.

Based on these assessments, the staff finds the two-phase mixture level model and rod heat transfer above the two-phase mixture level model during loop seal, boiloff, and recovery periods in M-RELAP5 acceptable for US-APWR SBLOCA analyses. The staff finds cladding heat-up is conservatively predicted for high pressure conditions by M-RELAP5 for US-APWR SBLOCA analyses.

4.4.1.3 ORNL/THTF High-Pressure Reflood Test

Following loop seal clearance, the two-phase mixture level in the core is recovered. Following the loop seal recovery, coolant boiloff in the core may occur due to the coolant loss through the break, such that the two-phase core mixture level may decrease again. The core mixture level is recovered when the SI rate exceeds the coolant loss through the break. When the reactor system pressure drops below the accumulator set-point the accumulators begin to inject into the cold legs and refill both the downcomer and the core.

Along with the core mixture level recovery in the core (core reflood), the fuel cladding temperature decreases as a result of improved cooling and finally drops to just above the saturation temperature when the cladding is rewet. The predictions of the improved cooling and rewet during the reflood phase are important to confirm core coolability during a SBLOCA.

A series of the high-pressure reflood tests were performed under conditions similar to those expected in a SBLOCA in THTF at ORNL. The objective of the reflood tests was to study bundle-rewetting (or quenching) behavior under conditions of varying system pressure, linear power, and flooding rate. These tests were used to assess the M-RELAP5 code applicability to the prediction of the core reflood behavior in a SBLOCA.

THTF has a 64-rod, full-length rod bundle heat transfer loop. The rod diameter and pitch are typical of a 17x17 PWR fuel assembly. The scaling of the facility is full length with prototypical PWR dimensions. The staff finds the ORNL/THTF acceptable for use for the assessment of M-RELAP5 for the prediction of SBLOCA core reflood behavior.

The tests assessed with M-RELAP5 include the following:

- Test 3.09.10O 3.88 MPa (562.75 psia).
- Test 3.09.10P 4.28 MPa (620.76 psia).
- Test 3.09.10Q 3.95 MPa (572.90 psia).
- Test 3.09.10R 7.34 MPa (1064.58 psia).
- Test 3.09.10S 7.53 MPa (1092.13 psia).

MHI concluded M-RELAP5 adequately predicted fluid conditions, such as fluid collapsed level during reflood, and predicted higher fuel rod surface temperature, showing M-RELAP5 conservatively predicted the rod heat transfer behavior during reflood. MHI concluded it is reasonable to apply M-RELAP5 to the simulation of reflooding phase of US-APWR SBLOCA analysis.

In its response to Request 8.1.3-1 [MHI03], MHI provided a comparison of the THTF experimental flooding mass flux with that expected under the typical SBLOCAs in the US-APWR.

The US-APWR flooding mass flux data were extracted from the calculations for the core recovery period of the 7.5-in and 1.0 ft^2 cold leg break accidents described in the design control document. The 7.5-in break generates the highest PCT during the “loop seal” period and the 1.0 ft^2 cold leg break is the limiting case and generates the highest PCT during the “boiloff” period. The flooding velocity (mass flux) for the 7.5-in break was around 70 $kg/s\text{-}m^2$ (14 $lbm/s\text{-}ft^2$) and around 100 $kg/s\text{-}m^2$ (20 $lbm/s\text{-}ft^2$) for the 1.0 ft^2 break. The flooding velocity range for the tests was from 80.80 $kg/s\text{-}m^2$ (16.55 $lbm/s\text{-}ft^2$) from Test 3.09.10P and 52.45 $kg/s\text{-}m^2$ (10.74 $lbm/s\text{-}ft^2$) from Test 3.09.10Q. The staff concluded the flooding velocity range in the tests adequately covered the expected values for the US-APWR SBLOCA.

MHI made several comments regarding the discrepancy between the measured and calculated initial collapsed liquid levels. In the M-RELAP5 analyses, boundary conditions were defined for the FRS thermal power, the inlet coolant flow rate and subcooling, and the exit pressure.

The initial FRS surface temperature and liquid level were obtained from a steady-state calculation. MHI concluded the initial FRS surface temperature was more important than the initial liquid level to simulate the reflood quench behavior because the temperature directly affects the onset of rewetting. Initial calculations with the measured flow and subcooling resulted in higher FRS surface temperature than measured. These higher initial temperatures would delay the rewetting and might make the basic models appear more conservative than they actually are. Therefore, the initial values of inlet flow and subcooling were adjusted in the steady-state calculation so that the calculated FRS surface temperature agreed with the measured temperature, resulting in the slight mismatch between the measured and calculated initial liquid levels. The staff accepts the initialization method used to match the M-RELAP5 FRS temperatures with the measured values when using these tests to assess the reflood model in M-RELAP5.

In its response to Request 8.1.3-8 [MHI03], MHI addressed the staff’s observation that the quench velocities for Test P and Test Q were similar for the tests (about 3 cm/sec (1.2 in/sec) for Test P and 2.8 cm/sec (1.1 in/sec) for Test Q), but they were substantially different for the M-RELAP5 analyses (about 2.4 cm/sec (0.9 in/sec) for Test P and 1.3 cm/sec (0.5 in/sec) for Test Q). This implied that the predicted quench velocities could be higher than the test values for some parameter ranges.

MHI stated that the test report described the average quench rate for Tests P and Q as 3.28 cm/s (1.29 in/s) and 2.78 cm/s (1.09 in/s), respectively. This difference was due to the difference in the flooding rate, 9.2 cm/s (3.62 in/s) for Test P and 5.9 cm/s (2.32 in/s) for Test Q. In addition, the test report pointed out that the heat transfer mode around the quench front affected the quench rate, based on ORNL investigations of the dynamic behavior of the measured histories in terms of quench and collapsed levels.

When the quench level is significantly above the collapsed liquid level, ORNL considered dispersed flow film boiling to be likely, while an inverted annular film boiling may be likely when the quench level is near or below the collapsed level. MHI provided comparisons between the measured quench level and collapsed level histories for Tests P and Q, indicating that dispersed flow film boiling may have occurred in Test Q but not in Test P. With dispersed flow

film boiling, like in Test Q, droplets can occur at the liquid level front, which would contribute to enhanced cooling above the liquid level before quench. This effectively decreases the vapor superheat and the fuel rod surface temperature in the uncovered-bundle region, and speeds up the quench rate even though the flooding rate was low. This was why the difference in the measured quench rates between Tests P and Q was relatively small, even though the flooding rates were significantly different from each other. The staff finds that the comparisons provided the explanation requested for the observed differences.

The ratio of quench-to-flooding rate from M-RELAP5 was smaller in the lower flooding case (Test Q) when compared to the measurements. In M-RELAP5, this can be attributed to the wall heat transfer model associated with the uncovered-bundle region in M-RELAP5. M-RELAP5 does not model an explicit droplet field and therefore does not consider the cooling effect of droplets on the heated wall and vapor just above the mixture level. MHI concluded the wall heat transfer for dispersed flow film boiling (Test Q) was conservatively represented by the vapor convection (single-phase vapor) model used in M-RELAP5, while the wall heat transfer for inverted flow film boiling (Test P) was reasonably modeled with the Bromley [TEC11] heat transfer correlation. Based on this discussion, the staff finds the M-RELAP5 vapor convection model conservative and acceptable for use in the US-APWR SBLOCA analyses. The staff finds the use of the Bromley correlation acceptable for use in the US-APWR SBLOCA analyses.

MHI also compared the collapsed and quench levels calculated by M-RELAP5 for Tests P and Q. MHI also showed the axial distributions of the applied heat transfer mode and the computed heat transfer coefficient from M-RELAP5 for these tests. The predicted cooling effect from M-RELAP5 was less than the measurement for Test Q, where the enhanced cooling effects (from droplets) were observed. It was found that the dryout heat transfer coefficient just above the quench front in Test Q was reduced to nearly half that of Test P, due to the heat transfer mode different from Test P. This showed that the enhanced cooling effect was conservatively underestimated in Test Q, resulting in the underestimation of quench rate in Test Q. The staff agrees the enhanced cooling effect is conservatively underestimated. A delay in quench results in a longer heat-up and a higher PCT.

MHI concluded M-RELAP5 tends to underestimate the heat transfer coefficient both for the low flow/dispersed flow film boiling and for the high flow/inverted flow film boiling, and to predict slower quench rates in comparison to the measurements. The staff finds the underestimation of the heat transfer coefficients and slower predicted quench rates acceptable and conservative for M-RELAP5 US-APWR SBLOCA analyses.

4.4.1.4 Flecht-Seaset Forced-Reflood Test

The M-RELAP5 models for the fuel rod heat transfer and rewet phenomena during the core recovery phase in the US-APWR SBLOCA were confirmed by the comparison with the ORNL/THTF high-pressure reflood test data. Those tests simulated representative small break core recovery and were conducted for a high pressure range of greater than 3.9 MPa (565 psia). However, the pressure decreases to less than 1 MPa (145 psia) during the core recovery phase for the 1.0 ft^2 cold leg break, which is the limiting US-APWR SBLOCA. The staff, in Request 7-16 [NRC04], requested that MHI provide comparisons of M-RELAP5 to confirm the conservatism of the heat transfer models under the lower pressure conditions typical of the larger break size. MHI provided its response in Reference MHI06. Therefore, M-RELAP5 was validated using the forced-reflood test data obtained in the Full-Length Emergency Core Heat Transfer for the Separate Effects and Systems Effects Tests (FLECHT-SEASET) program to confirm its applicability to the core recovery under the low pressure conditions [DAT04].

Three forced-reflooding tests, Runs 31504, 31701, and 32013, were selected by MHI to assess the applicability of M-RELAP5 for low pressure conditions. Run 31504 was the reference experiment, while Runs 31701 and 32013 corresponded to the higher reflooding and higher pressure experiments, respectively. Table 4 provides the experimental conditions for the selected tests. MHI compared the test conditions with the expected US-APWR condition in Reference MHI06:

Table 4. FLECHT-SEASET Test Data

	FLECHT-SEASET Run		
Parameter	31504	31701	32013
Pressure (psia) (MPa)	40 (0.28)	40 (0.28)	60 (0.28)
Inlet velocity (in/s) (cm/s)	0.97 (2.7)	6.10 (12.5)	1.04 (2.6)
Inlet subcooling (ºF) (°C)	144 (80)	141 (78)	141 (78)
Initial rod peak power (kW/ft)	0.7	0.7	0.7
Max initial temperature (ºF)(°C)	1507 (819)	1640 (893)	1555 (846)

Topical Report MUAP-07013-P (R2) Figures 8.1.4-2 through Figure 8.1.4-7 illustrates the comparisons of the heater rod temperatures at the 72-in and 96-in elevations, where the PCT occurred in the experiments. M-RELAP5 predicted a PCT higher than the test data for all three cases. M-RELAP5 also predicted a longer dryout period (rewet) than the test data. For the higher reflooding test, Run 31701, M-RELAP5 provided a slightly less conservative prediction than in the other test cases. Therefore, MHI concluded that the code conservatively predicts the fuel cladding heat-up for US-APWR SBLOCA low pressure conditions, as well as high-pressure conditions as demonstrated by using the ORNL/THTF test data.

The staff finds cladding heat-up is conservatively predicted by M-RELAP5 for US-APWR SBLOCA analyses including low pressure conditions, as well as high pressure conditions, as demonstrated by using the ORNL/THTF test data.

4.4.1.5 UPTF Full-Scale SG Plenum CCFL Test

Heat removal by the SG plays an important role in the SBLOCA when the break flow rate is small and the primary pressure remains higher than the secondary side pressure. Condensed water in the SG U-tube either accumulates in the SG U-tube and SG inlet plenum or flows back to the RV against steam flow. CCFL characteristics in the SG U-tube and the hot leg will affect core cooling due to the behavior of the condensed water in the SG U-tube.

Verification of the M-RELAP5 CCFL modeling in the hot leg region was conducted using the UPTF hot leg CCFL experiment [DAT05]. Since the UPTF hot leg SET is a full scale model, scaling was not an issue. The staff finds use of the UPTF acceptable for the evaluation of the M-RELAP5 CCFL model.

System pressure and flow conditions for the tests were as follows:

- System pressure: 3 bar, 15 bar

- Water flow rate: 30 kg/s

- Steam flow rate: 12 kg/s to 20 kg/s (six tests) conditions for 3 bar
  24 kg/s to 40 kg/s (ten tests) conditions for 15 bar

Results of M-RELAP5 analysis with the CCFL correlation showed that the characteristics of the water downflow rate against the steam upflow rate was reasonable for both the 3 bar (0.3 MPa, 43.5 psia) and 15 bar (1.5 MPa, 213 psia) conditions. The CCFL parameters were derived for a large diameter pipe from the UPTF CCFL test data.

MHI concluded the M-RELAP5 CCFL model for the behavior of the hot leg and the SG plenum is applicable for the US-APWR. The staff agrees with this assessment.

In its response to Request 8.1.4-3 [MHI03], MHI explained the criteria used for selecting the use of either the “big” pipe or “small” pipe form of the CCFL correlation. The M-RELAP5 documentation indicated that for large pipes the fluid surface tension is important (Kutateladze number) and for small pipes the length scale does not depend on surface tension. However, there is no statement on the size of the pipe where the transition occurs.

The Kutateladze number (depending on steam flow rate) shows that the no water penetration point increases with dimensionless pipe diameter and approaches a constant value of about 3.2 for a dimensionless pipe diameter greater than about 60. MHI derived a table for the physical pipe diameter as a function of pressure from this value. Based on this table, the CCFL parameters derived from the UPTF CCFL test data are applied to the hot leg nozzle of the SG inlet plenum. The derived table was applied to the inlet of the U-tubes in the SGs using a hydraulic-equivalent diameter for the length-scale and it was determined that the CCFL parameters based on the Hewitt and Wallis correlation are appropriate for this region. The staff finds the CCFL parameters used for the hot leg nozzle of the SG inlet plenum and for the SG U-tube inlet acceptable.

In its response to Request 8.1.4-5 [MHI04], MHI explained the rationale for the number of nodes in the different sections of the UPTF M-RELAP5 model and how this nodalization compared to nodalization of similar sections for the US-APWR.

The nodalization for the hot leg in the UPTF M-RELAP5 model was the same as that used in Reference DAT18. In the US-APWR M-RELAP5 model, both the horizontal part and the riser part of the hot legs were represented. Sensitivity analyses were performed by MHI to investigate the effect of nodalization. MHI concluded the number of nodes does not significantly affect the results and the application of the CCFL model does not depend on the number of nodes. The staff agrees with this assessment.

In its response to Request 8.1.4-8 [MHI06], MHI addressed potential for CCFL at the upper core plate and neutron reflector holes.

CCFL models were considered at locations where the phenomenon could significantly affect core thermal-hydraulics. MHI performed sensitivity studies with and without the CCFL models at each location. The Bankoff model for a perforated plate [TEC12] was applied to the upper core plate and the Wallis model for a sharp-edged vertical pipe [TEC13] was applied to the neutron reflector exit-holes.

The sensitivity analyses were performed for the 7.5-inch cold-leg break where the “loop seal” PCT occurred and for the 1.0 ft^2 break where the boiloff PCT occurred. No significant effects were observed on the collapsed liquid levels in the pressure vessel and on the PCT except for the quench time in the 1.0 ft^2 break case where quench occurred a little faster with the CCFL models.

The least conservative PCT deviation between the CCFL model on and off was +0.6 ºF (0.33 K) for the 7.5-inch break, neutron reflector case, which is a small increase in the PCT with the CCFL model on.

The staff agrees with the applicant's findings that the CCFL model has been appropriately applied to the US-APWR SBLOCA EM.

The effect of pressure on CCFL was recently reported in NURETH-13 by the Dresden group [TEC14]. The hot leg geometry was simulated using a rectangular duct of 5 cm (2 in) in width and 25 cm (9.8 in) in height. High pressure steam-water experiments were conducted at 15 bar, 30 bar and 50 bar. The CCFL data for the different pressures correlated reasonably well with Kutateladze number. However, it is recognized that the Kutateladze number for the water down-flow rate at a given steam flow rate tends to increase with pressure. MHI concluded this tendency means that the UPTF correlation derived below 15 bar gives conservative results under higher pressure because more water accumulates around the SG inlet plenum when using the UPTF correlation than would be expected under higher pressure. The additional water accumulation around the SG inlet plenum causes a lower liquid level in the core during the loop seal clearing period, which increases the likelihood of core dryout.

Since the CCFL correlation strongly depends on flow-path geometry, the Kutateladze relationship by the Dresden group using the rectangular geometry cannot be applied directly to the US-APWR. However, the qualitative tendency of the effect of pressure on the liquid down-flow is considered to be relevant to the US-APWR. Therefore, the use of the CCFL correlation derived from the UPTF data is considered to be conservative in the M-RELAP5 SBLOCA analyses. The staff agrees with this assessment.

In its response to Request 8.1.4-11 [MHI08], MHI provided a quantitative justification for the use of the pre-determined CCFL parameter values for the US-APWR SBLOCA analyses, including the selection of the Kutateladze number, with a value of 3.2.

The hot leg inner diameter for the UPTF facility is 0.75 m (2.46 ft) and 0.787 m (2.58 ft) for US-APWR. Since the Kutateladze number for zero penetration of water approaches a constant value with increasing inner diameter, MHI concluded the CCFL parameters derived from the UPTF data are applicable to the US-APWR because the diameter of 0.75 m (2.46 ft) is large enough.

In its the response, MHI did not consider the actual minimum flow areas of the hot legs. The difference of the minimum flow area, where the CCFL would occur, between the US-APWR hot leg and the UPTF hot leg is about 23 percent and is not small enough to be ignored. MHI was asked to answer the original request quantitatively, as requested, considering the difference in the minimum flow area. MHI was also asked to address the applicability of the CCFL correlation used at the minimum flow area at the “Hutze” in the UPTF. MHI provided its response to Request 8.1.4-11-1 in Reference MHI12.

The empirical Kutateladze number, from the UPTF data, was derived by using the flow area of 0.442 m^2 (4.76 ft^2) for the round pipe with a 0.75 m (2.46 ft) inner diameter. If the smaller area of 0.397 m^2 (4.27 ft^2) had been used to account for the "Hutze," (where the ECC injection channel, called ''Hutze'', was installed) the coefficient *c* in the correlation would be larger than the value used for the plant calculation by about 6 percent and the coefficient *m* would be the same.

Applying the larger *c* value to the US-APWR would cause the water downflow rate to become larger at the same steam upflow rate. This means that the CCFL correlation which accounts for the "Hutze" tends to give non-conservative results because the amount of water accumulation due to the CCFL is decreased and the driving head depressing the core liquid level decreases. The treatment used to derive the CCFL correlation is considered by MHI to be adequate and conservative.

The staff finds the M-RELAP5 CCFL correlation used in the hot leg region acceptable and conservative for use in the US-APWR SBLOCA analyses.

4.4.1.6 Dukler Air-Water Flooding Test

Heat removal by the SG plays an important role in the SBLOCA when the break flow rate is small and primary pressure remains higher than secondary side pressure. Condensed water in the SG U-tube either accumulates in the SG U-tube and SG inlet plenum or flows back to the RV against steam flow. CCFL characteristics in the SG U-tube and the hot leg will affect core cooling through behavior of the condensed water in the SG U-tube.

Verification of the CCFL modeling in relatively small diameter pipe like a SG U-tube by the M-RELAP5 was evaluated using the Dukler Air-Water Flooding Test [DAT06].

In its response to Request 8.1.5-1 [MHI03], MHI provided a scaling discussion of the Dukler Air-Water Flooding test facility for comparison to the US-APWR SG tubes.

1. Tube diameter: CCFL in the SG tubes depends on the interaction between the liquid film condensed within the tube and the upward steam flow. This implies the impact of the tube end geometry will have little influence on the CCFL. Therefore, MHI selected the Hewitt and Wallis correlation since the tube end effects are expected to be minimal. MHI considers the J* scaling to be applicable to a small-scale pipe as discussed in the response to Request 8.1.4-3 [MHI03]. The Dukler experimental data were plotted against the CCFL correlation as shown in Topical Report MUAP-07013-P (R2) Figure 8.1.6-4, "Comparison of Calculated and Measured Results Using the Wallis Flooding Correlation Constants." The plot indicates the model correlates well to the data irrespective of the tube diameter of 3/4 in or 5/4 in. Since a tube diameter of ¾ inches is near the US-APWR U-tube diameter, MHI concluded the Hewitt and Wallis correlation is applicable to the US-APWR U-tube inlet region.

2. Tube length: The phenomena restricting the downward liquid flow rate in SG tubes is considered to be governed by those near the bottom of the tubes where the steam and condensed liquid flows are maximized. MHI believes the effect of tube length is unlikely to be important for this situation. Several tests were examined in the original correlation paper [TEC13] and the effect of length was not found to be a parameter affecting the CCFL.

3. Tube wall material: The effect of wall friction is considered to be smaller than the interfacial friction. MHI did not find any experimental studies on the wall friction effect on CCFL. The coefficient “c” used in the MHI evaluation of the ROSA/LSTF is the same as that used to evaluate the Dukler data. This implies the effect of tube wall material is not significant because the ROSA/LSTF uses stainless-steel tubes and the Dukler experiment Plexiglas. In the penetration region, the wall friction might have some effect because the water down flow rate in the Dukler experiment tends to be larger than the data used to develop the correlation.

4. Fluid combination: A study on counter-current two-phase flow [TEC14] revealed that the difference between fluid combinations (air/water versus steam/water) can be scaled by the J* parameter. The ROSA/LSTF tests investigated the steam flow rate giving zero water penetration at the bottom of the SG tubes [DAT19], and the same coefficient “c” was used. MHI concluded that these results support use of the same coefficient set for the US-APWR SBLOCA analyses.

The staff finds the scaling assessment of the Dukler Air-Water Flooding test facility acceptable, and use of the facility for the evaluation of the CCFL model used for the SG U-tubes in the US-APWR SBLOCA analyses is also acceptable.

The analysis of the Dukler Air-Water Flooding Test was conducted using M-RELAP5 with CCFL parameters proposed by Hewitt and Wallis. The analysis results showed good agreement with the test data. This analysis demonstrated that M-RELAP5 with the CCFL parameters proposed by Hewitt and Wallis is applicable to simulation of CCFL behavior of small diameter pipe such as SG U-tube in the US-APWR. The calculated water downflow rate was 30 percent smaller than the test data, on the average, which is conservative because in the plant application it will result in less liquid return to the RV. The staff finds the use of the Hewitt and Wallis parameters in the CCFL model for the SG U-tubes acceptable for the US-APWR SBLOCA analyses.

In its response to Request 8.1.5-2 [MHI03], MHI provided a comparison of the test pressure, temperature and flow rates (both water and air) with those expected at the SG U-tube uphill side during the loop seal clearing period, and discussed why these tests are applicable to the loop seal period of a SBLOCA.

Based on the discussion provided in its response to Request 8.1.5-1 [MHI03], MHI concluded the CCFL correlation can account for the differences (scaling effects) in the configuration and fluid combination between the Dukler test facility and the US-APWR. Furthermore, the CCFL correlation is an important contribution during the loop seal clearing period. A comparison with the CCFL correlation to test data from the ROSA/LSTF was also provided to show that CCFL is important prior to loop seal clearing. CCFL governed the down flow rate in the loop seal formation-clearance period until the time the loop seal clears.

In its response to Request 8.1.5-5 [MHI03], MHI discussed the test coverage for all flow regimes since reflood/reflux flow would be affected by the flow regime.

In the Dukler flooding test, annular countercurrent flow was mainly investigated and a slugging flow was reported under a low air upward flow rate after flooding occurred. Basically, the same flow regime is predicted in the M-RELAP5 analyses although the predictive correctness of the flow regime boundary is not clear due to lack of experimental information especially on the axial variation of the flow regime. The amount of liquid accumulation within the uphill side of SG U-tubes is one of most important values affecting the core liquid level. The CCFL characteristics

and the flow regime predictions affect the value. The former subject was investigated and the correlation for flooding velocities in an air-water system was confirmed to apply to the actual conditions as stated in the responses in Request 8.1.5-1 and 8.1.5-2. The latter one (flow regime predictions) was indirectly evaluated with the ROSA-IV/LSTF analysis through comparisons of differential pressures along the uphill side of SG U-tubes. Reasonable agreements were obtained on the differential pressure shown in Topical Report MUAP-07013-P, Figures 8.2.1-26 and 8.2.1-27.

The liquid accumulation was shown to be adequately predicted and the good predictions for the differential pressures indicated no significant problems due to flow regime predictions.

In its response to Request 8.1.5-7 [MHI04], MHI provided nodalization sensitivity studies for Dukler tests.

Sensitivity analyses were performed by MHI with finer node and no significant differences were observed.

Since the node length of the SG U-tubes in the US-APWR is coarser than that for the test calculation, an additional sensitivity analysis was performed using a longer node length. The effect of this change was negligibly small. Based on these studies, the staff finds the nodalization of the SG U-tubes in the US-APWR SBLOCA analyses acceptable.

The staff finds the M-RELAP5 CCFL correlation parameters used for the SG U-tubes acceptable for use in the US-APWR SBLOCA analyses.

4.4.2 Integral Effects Tests

4.4.2.1 ROSA-IV/LSTF Small Break (5 percent) LOCA Test (SB-CL-18)

The purpose of this ROSA-IV/LSTF calculation was to validate the M-RELAP5 code performance to predict the following high ranked processes and phenomena identified in the US-APWR SBLOCA PIRT: core dryout, post-CHF heat transfer, rewet, core mixture level, water hold up in SG primary side, condensation drainage to the SG inlet plenum, SG primary and secondary heat transfer, water level in the SG outlet piping, loop seal formation and clearance, and downcomer mixture level. The 5 percent break is equivalent to a 6 in. break (SBLOCA) in a four-loop PWR.

The nodalization for the test facility and the US-APWR SBLOCA were substantially different: notably, the core bypass model, the number of nodes in the loop seals, and the number of nodes in the SG U tube. These differences could affect the analysis results, especially for the SG primary and secondary heat transfer, the loop flow rates (due to different friction factors) and loop seal clearance (number of nodes in the loop seal region). In response to Request 8.2.1-5 [MHI03], MHI explained the reasons for using different nodalization for the test facility and the plant, and discussed the sensitivity of the calculations with respect to nodalization.

The ROSA test facility does not have an in-core bypass region such as the core thimble region or the neutron reflector that are present in the US-APWR. Therefore, the ROSA core model does not have a core bypass region model.

The loop seal pipe diameter in ROSA-IV is smaller than that of the US-APWR. To conserve the cell length-to-diameter (L/D) ratio of the crossover leg noding, a finer cell size was adopted for

the ROSA-IV noding. The nodalization of the crossover leg is similar to the US-APWR model based on the L/D ratio. To keep the level of detail consistent between the U-tube noding and the crossover leg noding, a finer cell size was also adopted for the U-tube.

Crossover leg and the SG U-tube nodalization sensitivity studies were performed with the US-APWR model, and reported in Technical Report MUAP-07025-P [MHI24]. The number of cells in the crossover leg and the U-tubes were approximately doubled compared to the base model.

The results from the sensitivity studies for the RCS pressure transient for the sensitivity case agrees with that of the base case. The results showed moderate sensitivity in the calculating of the collapsed liquid level and the cladding temperature during the loop seal clearance period. MHI also noted in US-APWR analysis, cladding heat up during the loop seal clearing period is not large. MHI uses the more conservative model for the US-APWR SBLOCA EM.

The staff finds the use of the more conservative noding model used for the US-APWR SBLOCA analyses acceptable.

Based on the discussion provided, the staff also finds the noding model used for the ROSA facility is acceptable for evaluating M-RELAP5 capabilities related to core dryout, post-CHF heat transfer, rewet, core mixture level, water hold up in SG primary side, condensation drainage to the SG inlet plenum, SG primary and secondary heat transfer, water level in SG outlet piping, loop seal formation and clearance, and downcomer mixture level.

Base Case Analysis

The break flow and secondary pressures were input as boundary conditions so that the validation could focus on the code's ability to calculate important RCS phenomena such as natural circulation, liquid holdup in the U-tubes, loop seal clearing, core uncovery, and core heat-up without the effect of differences in the experimental system, such as break geometry, steam generator heat loss, and steam generator valve leakage. The break flowrate was reduced (by 35 percent) to make the primary system depressurization agree with test data during the single-phase vapor break flow region (after about 145 sec).

The M-RELAP5 predicted core water level dropped earlier than in the test (300 sec for the analysis versus 400 sec for the test) during the boiloff period. MHI attributed this to more liquid remaining in the crossover legs in the analysis. However, M-RELAP5 showed the water holdup was larger during the period preceding this difference, and the difference was decreasing at 300 seconds. This observation seemed contradictory to the explanation. MHI addresses this apparent inconsistency in its response to Request 8.2.1-6 [MHI04].

An additional sensitivity analysis was done to increase the liquid mass releasing from the crossover leg to the core by increasing the interfacial drag at the crossover leg uphill side. The liquid mass in the crossover leg uphill side, upper plenum, and the core, were all nearer to the test data. The difference with the test data became smaller; however, there was still a difference of several tens of seconds in the time when the reactor core differential pressure began to decrease. The prediction of the liquid mass release from the upper head affects the core liquid mass; however, there were no test data that show how much liquid remains in the upper head. The measurement accuracy of the break flow rate and the amount of liquid remaining in the hot legs also affect the amount of liquid in the RV.

In its response to Request 8.2.1-6 [MHI04], MHI explained the earlier decrease in the core collapsed liquid level in the M-RELAP5 simulation resulted from inaccuracies in the predicted liquid mass being released from the upper head to the downcomer. MHI obtained this conclusion by performing a sensitivity study by increasing the number of spatial meshes for the upper head region to see the steam separation effect. MHI showed an increase in the upper plenum water level, where the liquid phase was expected to occupy the lower nodes and flow to the top of the downcomer during the boiloff period. However, the amount of the increase in the upper plenum water level was small compared to the measured data (Figure RAI-8.2.1-6.7).

In its response to Request 8.2.1-16 [MHI08], MHI discussed the possibility that uncertainties in the flow area of the spray nozzle between the downcomer and the upper head could be the cause of this inaccuracy in the prediction.

Reference DAT20 provides the data relevant to the pressure drop at the downcomer spray nozzle, and the estimated uncertainty was about 10 percent.

In addition, it is found that the bypass flow rate was measured after several major transient tests in the ROSA-IV/LSTF facility, and that the bypass flowrate fraction probably shifted from 0.3 percent to 0.4 percent when the SB-CL-18 test was performed. The 0.3 percent bypass fraction used for the SB-CL-18 test analysis was determined from the data measured during the facility shakedown test.

Based on these observations, MHI performed new sensitivity calculations for the spray nozzle flow area such that the bypass fraction varied from 0.2 percent to 0.4 percent. The calculated core differential pressures were compared to the base case in Figure RAI-8.2.1-16.1. The calculations were not sensitive to the uncertainty in the bypass flow area (bypass fraction). The M-RELAP5 predictions were conservative in the sense that the core liquid level was underestimated between 300 and 400 seconds. The total system mass is the same in the data and calculations so the lower mass in the core region must be due to a difference in the mass distribution within the system. As there are no data on mass distribution except for the core region, it is not possible to know where the liquid mass that is not predicted to be in the core region is located. The applicant established that this difference does not affect the PCT results. Therefore, the staff finds MHI's response to RAI 8.2.1-16 acceptable

In its response to Request 8.2.1-8 [MHI08], MHI explained why different CCFL values were used in this IET assessment, as compared to those used in the US-APWR.

The CCFL parameters from the UPTF data were applied to US-APWR and their validity was described in Response of RAI 8.1.4-11 [MHI08]. The CCFL parameters based on a study by Tien et al [TEC15] were applied to the ROSA-IV/LSTF analysis. The parameter set was changed because the water accumulation in the ROSA-IV/LSTF SG inlet plenum was better predicted with the Tien model. MHI conducted a sensitivity analysis for ROSA-IV/LSTF SB-CL-18 where the Tien model was replaced by a J^* correlation developed using a simulated hot leg geometry. The J^* correlation from Richter et al. [TEC16] was used because the inner diameter of 0.203 m is close to the ROSA value of 0.207 m.

Figures RAI-8.2.1-8.1 and RAI-8.2.1-8.2 [MHI08] showed the comparison of differential pressures in the broken loop SG inlet plenum and in the one intact loop, respectively. Differences began to appear after about 100 seconds, which corresponded to the start of the loop seal period, and the differential pressure decreased more rapidly with the Richter correlation. The Tien model gave better predictions and was selected for the final assessment.

The staff finds the use of the Tien model in the assessment acceptable because it provided a better simulation of the data.

The Kutateladze number for zero penetration of water approaches a constant value with increasing inner pipe diameter (see Response 8.1.4-3 in Reference MHI03). The response was based on vertical pipes and the Kutateladze number for zero penetration of water is about 3.2. However, the Kutateladze number for zero penetration of water for the hot leg geometry is different from that for the vertical pipe and is about 7.18 for the UPTF. Since the CCFL behavior is strongly dependent on the geometry, the effect of diameter on the Kutateladze number for zero penetration of water is considered to be different for the vertical pipe and the hot leg geometry. Therefore the *c* value was increased, based on scale, to the value for the UPTF.

It was understood from MHI's response to Request 8.2.1-8 [MHI08] that the CCFL parameter values depend on the geometry of the test facility. Since the geometry of US-APWR is different from UPTF and ROSA-IV/LSTF, MHI's response to RAI 8.2.1-8-1 [MHI12] explained why the selection of the CCFL parameters from the UPTF data at the junction between the hot leg and the SG plenum, and the CCFL parameters from ROSA-IV/LSTF at the junction of the crossover leg uphill side, gave meaningful or conservative results for the US-APWR SBLOCA analysis.

The applicability of the UPTF hot leg CCFL parameters to the US-APWR was described in the MHI response to RAI 8.1.4-11 [MHI08] and RAI 8.1.4-11-1 [MHI12].

Even though the inner diameter of the hot leg in the UPTF is about 5 percent less than the inner diameter of the US-APWR, both diameters are large enough that the Kutateladze number should be acceptable for predicting zero water penetration. The Kutateladze number derived from the UPTF is considered to be high. Even though the Kutateladze number for zero water penetration tends to increase with the diameter for the hot leg, the CCFL parameters from the UPTF are considered to give a conservative result for the US-APWR as discussed in MHI's response to RAI 8.1.4-11-1.

The staff agrees with the use of geometry-based CCFL parameters in the M-RELAP5 evaluation of test facilities.

MHI stated, "Break flowrate of M-RELAP5 calculation is adjusted to test data (Figure 8.2.1-15), as a result, primary pressure drop behavior agrees with test data (Figure, 8.2.1-16). Signal timings agree with test data (Table 8.2.1-6). Secondary pressures are also adjusted to test data (Figures 8.2.1-13 and -14). Primary pressure and secondary pressures of M-RELAP5 calculation agree with test data, and as a result, M-RELAP5 capability to predict SG primary and secondary heat transfer is good."

It is understandable that the primary and secondary system pressures show good agreements between the prediction and the measurement because the primary system pressure was predicted with the adjusted break flow and the secondary system pressure was also adjusted to the test data. However, this is not the manner in which the plant calculations will be performed, i.e. break flow and secondary pressure will not be specified as boundary conditions. In its response to RAI 8.2.1-12 [MHI07], MHI explained how this assessment establishes the ability of M-RELAP5 to predict the plant response for SBLOCA events by performing a sensitivity study that mechanistically modeled the main steam isolation and relief valves.

The SG primary and secondary side heat transfers were important phenomena during the blowdown, natural circulation, and loop seal clearance phases in the US-APWR SBLOCA PIRT.

The heat transfer through the SG U-tubes was validated by comparing the integral of steam mass discharged out the main steam isolation and relief valves between the calculation and measurement.

MHI showed M-RELAP5 overestimated the steam released from the SGs, as shown in Figure RAI-8.2.1-12.3, “Integral of SG Secondary Outlet Steam Mass”, however, the difference between the calculation and measurement of the total discharge was less than 10 percent. This overestimation was judged by MHI to have a small impact on the primary system pressure response as shown in Figure RAI-8.2.1-12.4, “Pressurizer pressure.” In addition, Figure RAI-8.2.1-12.5, “Core Differential Pressure,” showed that the core liquid level (differential pressure) in the sensitivity calculation agreed with that obtained by the base model.

MHI concluded that the code assessment results presented in the topical report were sufficiently meaningful, even though the secondary system pressure was imposed as a boundary condition.

In its response to Request 8.2.1-12-1 [MHI12], MHI explained how the assessment of ROSA-IV/LSTF SBLOCA (5 percent) test establishes the ability of M-RELAP5 to predict the US-APWR response for SBLOCA events in light of specifying important parameters as boundary conditions.

An additional assessment using the break flow model was performed for the ROSA-IV/LSTF. In this calculation, the secondary system behaviors were also simulated. The CHF multiplier for low flow and high void fraction conditions was also used in this calculation. The results of this assessment are provided in 8.2.1.9, “Sensitivity Calculation with Simulated Secondary System and Break Flow Behaviors,” in Topical Report MUAP-07103-P [MHI23].

The primary system depressurized after the break initiation until the pressure was equalized at a level slightly above the secondary system pressure, through the natural circulation phase to the loop seal clearance. The primary and secondary pressure responses agreed well with the measured values from the break initiation through the loop seal clearance. Almost all of the heat transferred from the primary side to the secondary side through the SG was used to generate vapor in the SG and most of the generated vapor was released through the main steam line or the steam line relief valves. These agreements of the primary/secondary pressure and the integrated vapor mass release from the SG showed that M-RELAP5 is able to predict the SG primary to secondary heat transfer.

The predicted PCTs were higher than the measured values. This confirmed M-RELAP5 conservatively predicts the PCTs when the analysis method applied to US-APWR SBLOCAs concerning the break flow and secondary system modeling is used for the code assessment using the ROSA experimental data. For this analysis, the 20 percent conservatism for the decay heat was not taken into account.

Sensitivity Studies

Additional sensitivity studies were performed to evaluate various aspects of the M-RELAP5 models and nodalization for comparison to the base case analysis, where the break flow and secondary pressures were input as boundary conditions.

Sensitivity-1 Calculation for CHF Multiplier

A sensitivity calculation was performed to validate the additional multiplier to the CHF described in Topical Report MUAP-07013-P Section 7.1.7.2.

There were some heater rod data in the upper portion of the experiment that indicated water accumulated in the upper plenum region and partially flowed down into the core region. Liquid distribution effects, like this, are modeled as CHF multiplier in M-RELAP5. However, the base case results indicated that this CHF multiplier was not enough for ROSA-IV/LSTF SBLOCA analysis. Therefore, the sensitivity calculation with additional CHF multiplier was performed.

MHI concluded that M-RELAP5, with the normal CHF multipliers, can simulate the overall hydraulic behavior during the loop seal period and can predict the heat-up behavior of the average heater rod due to core uncovery, and M-RELAP5, using conservative conditions for dryout (with the additional multiplier), can also predict the heat-up behavior in the upper core region. The staff agrees with this conclusion.

Sensitivity-2 Calculation for Decay Heat Multiplier

A sensitivity calculation, where the core decay heat was increased by 20 percent (Appendix K requirement for plant analyses), was performed to quantify its impact on PCTs.

The increased decay heat increased the amount of vapor generation in the core and the core liquid level did not completely recover after loop seal clearing. As a result, the onset of the heat-up during the boiloff period occurred earlier and conservative rod temperatures were predicted.

Three additional sensitivity analyses were performed to investigate the difference in core water level between the base case and test data.

Sensitivity-3 Calculation for Upper Head Nodalization

An upper head nodalization sensitivity calculation was performed to examine the effect of steam separation in the upper head using a finer nodalization in the upper head. The amount of the released liquid from the upper head to the core increased, and the liquid mass in the core, the downcomer, and the upper plenum increased. This sensitivity study confirmed that the use of the US-APWR upper head model is conservative with respect to the predicted PCT.

Sensitivity-4 Calculation for Crossover leg Interfacial Drag

A crossover leg interfacial drag sensitivity calculation was performed to increase the liquid mass released from the crossover legs to the core by increasing the interfacial drag on the uphill side of each crossover leg. Sensitivity-4 includes the Sensitivity-3 change. MHI concluded the prediction of liquid mass in the crossover leg was improved, and liquid mass release from the crossover legs increased. The core and upper plenum liquid mass were nearer to the test data. The difference with the test data became smaller in Sensitivity-4. However, there was still a difference of several tens of seconds in the time when the reactor core differential pressure began to decrease. The prediction of liquid mass release from the upper head may affect the core liquid mass; however, there were no test data that show how much liquid remains in the upper head. The results of an inquiry by MHI to the organization that performed the SB-CL-18 test indicated there was an error of several percent in the measured break flow rate. The difference of the differential pressure in the upper plenum and the reactor core can be explained if there is an error of about 3 percent.

Sensitivity-5 Calculation for Break Flowrate and Hot-leg Interfacial Drag

Sensitivity-5 decreased the break flowrate by 3 percent prior to loop seal clearing and adjusted it to match the depressurization after loop seal clearing. MHI found that no liquid mass remained in the hot legs after 100s in the base case. Therefore, the liquid mass in the hot legs was increased by increasing the interfacial drag at the hot legs in the Sensitivity-5. Sensitivity-5 included the Sensitivity-4 changes. The liquid mass of the hot legs was nearer the test data, and the liquid mass in the upper plenum and the core were also nearer the test data. The pressurizer pressure in the blowdown phase agreed with test data even though the break flow rate was decreased by 3 percent. The upper plenum water level increased in comparison with Sensitivity-4 and the base case, because of the additional flow of water from the hot leg after loop seal clearance.

These sensitivity studies showed that various mechanisms affect the prediction of core water level during the boiloff period. The liquid release from the upper head is particularly important for predicting the core water level during the boiloff period. However, the sensitivity analyses show that the core and downcomer liquid mass was smaller than the test data even in these sensitivity analyses.

MHI concluded M-RELAP5 predicts a lower vessel mass inventory as compared to the measurement, regardless of the uncertainty in the upper head mass. Sensitivity-1 model was the most conservative with respect to the predicted PCT, and is consistent with the modeling approach used in the US-APWR. The staff agrees with this conclusion.

MHI concluded the conservatism of M-RELAP5 for US-APWR SBLOCA analysis was confirmed by this calculation, which was performed in the same manner as the plant analysis, and which includes the calculations of the break flow and secondary system behaviors. MHI also concluded M-RELAP5 adequately predicted the SG primary to secondary heat transfer in this calculation. The staff agrees with these conclusions.

Based on the analysis of the ROSA/LSTF SBLOCA (5 percent) test (SB-CL-18), the staff finds M-RELAP5 can adequately predict core dryout, post-CHF heat transfer, rewet, core mixture level, and downcomer mixture level phenomena occurring in the boiloff and core recovery phases in US-APWR SBLOCA.

4.4.2.2 ROSA-IV/LSTF SBLOCA (10 percent) Test (SB-CL-09)

The calculation of the ROSA-IV/LSTF test SB-CL-09 was used to validate the ability of the M-RELAP5 code to predict the following high-ranked phenomena in the US-APWR SBLOCA PIRT: core dryout, post-CHF heat transfer, rewet, core mixture level, water hold up in SG primary side, condensation drainage to SG inlet plenum, SG primary and secondary heat transfer, water level in SG outlet piping, loop seal formation and clearance, and downcomer mixture level. A 10 percent break in ROSA-IV/LSTF is equivalent to about a 10.5 inch break in the US-APWR.

Time-dependent tables, which were constructed from the experimental data, were used to represent the core power and pump coast-down curves. The break flow was explicitly simulated with the Moody critical flow model with an atmospheric boundary condition. The offtake/pullthrough model, described in Section 4.2.4, “Break Pull-Through Model,” of this SE, was applied. The secondary system pressure behavior was also explicitly simulated by

modeling the main steam isolation and steam relief valves with the imposed boundary condition for the feedwater flow following the reactor trip.

CCFL parameters were applied at SG U-tube inlet, hot leg, and crossover leg uphill side. The Hewitt and Wallis CCFL parameters were used for the SG U-tube inlet. The CCFL parameters for the hot leg were determined from the validation results of Dukler air-water flooding test analysis. The CCFL parameters for the crossover leg uphill were based on Tien [TEC15].

The M-RELAP5 prediction of the primary pressure transient agreed with test data until 75 seconds, and after 75 seconds, the depressurization is faster in the M-RELAP5 prediction. The M-RELAP5 prediction of the secondary pressure transient also agreed with test data.

The break flow was calculated with the Moody critical flow model, as is used in the US-APWR licensing calculation. Overall, M-RELAP5 provided a reasonable prediction of the break flow rate until the initiation of the accumulator flow at 160 seconds. Taking account of the faster depressurization in the M-RELAP5 calculation after 75 seconds, the calculated mass flow is expected to be higher than the actual value. M-RELAP5 over-predicts the break flow rate after 160 seconds because the accumulators start injecting safety coolant earlier in the calculation. The start of accumulator injection was earlier in the M-RELAP5 because the RCS depressurization after 75 seconds was faster in the M-RELAP5 calculation.

M-RELAP5 underestimated the core differential pressure except for a brief part of the loop seal phase. M-RELAP5 did not calculate the rapid core recovery after core power trip at 111 seconds because the M-RELAP5 heat transfer model is conservative for the evaluation of the quench behavior, and the vaporization of liquid in the core continues for more than 100 seconds after the trip in the M-RELAP5 calculation. The heater rods rewet and quench before the core power trip at the lower elevations, and within about 40 seconds after the core power trip at higher elevations in the test.

M-RELAP5 predicted the SG inlet plenum differential pressure reasonably well until 150 seconds. M-RELAP5 calculated an earlier decrease in the SG U-tube downhill side differential pressure transients. Loop seal clearance was a little earlier in the M-RELAP5 prediction. The residual amount of liquid was larger than the test data in the uphill side of the intact loop (loop-A) after loop seal clearance. MHI concluded M-RELAP5 predicts the overall behavior of loop seal clearance in this experiment.

M-RELAP5 overestimated the downcomer differential pressure until 100 seconds, and then slightly under-predicted until 160 seconds. After 160 seconds, accumulator injection starts in the prediction, and M-RELAP5 again over-predicts. MHI concluded the overall response in the downcomer liquid level predicted by M-RELAP5 agreed reasonably well with the measurement.

The differential pressure in the upper plenum increased after the core was completely quenched in both the test and the calculation. This increase occurs at about 160 s in the test and at about 215 s in the calculation. M-RELAP5 predicted that the core quenches more slowly than in the test.

At the upper elevation of the heater rod, the initiation of heatup was slightly later in the prediction. At the other elevations, the initiation of heatup was conservatively or adequately predicted. The rewet was later at all elevations of the heater rod in the prediction. Except for the upper elevations of the heater rod, the predicted heater rod surface temperatures were higher than the measured ones because the post-CHF heat transfer model in M-RELAP5 is

conservative and the predicted mixture level was lower than the measured one. The predicted and measured peak heater rod surface temperatures were 1423 F (1046 K) and 1215 ºF (930 K), respectively.

M-RELAP5 well predicts the 10 percent cold leg break transient. The results are similar to those obtained previously for the 5 percent cold leg break test. The calculated heater rod temperatures are higher than the measurements, demonstrating the ability of M-RELAP5 to predict the PCT conservatively for SBLOCAs with larger break sizes.

Based on the analysis of the ROSA/LSTF SBLOCA (10 percent) test (SB-CL-09), the staff finds M-RELAP5 can adequately predict core dryout, post-CHF heat transfer, rewet, core mixture level, and downcomer mixture level phenomena occurring in the boiloff and core recovery phases in US-APWR SBLOCA. It was also confirmed that the US-APWR SBLOCA EM is conservative with respect to the predicted PCT.

4.4.2.3 ROSA/LSTF SBLOCA (17 percent) test (IB-CL-02)

The calculation of ROSA/LSTF test IB-CL-02 was used to validate the ability of the M-RELAP5 code to predict the following phenomena ranked of high importance in the SBLOCA PIRT: core dryout, post-CHF heat transfer, rewet, core mixture level, and downcomer mixture level. Test SB-CL-18 simulated a postulated break size which corresponds to a 1.0 ft^2 break in the US-APWR. Because of the rapid depression in the primary system, the important phenomena occurring in the boiloff and core recovery phases in US-APWR SBLOCAs could be identified in the experiment.

Time-dependent tables, which were constructed from the experimental data, were used to represent the core power and pump coast-down curves. The break flow was explicitly simulated with the Moody critical flow model with an atmospheric boundary condition. The offtake/pullthrough model, described in Section 4.2.4, “Break Pull-Through Model,” of this SE, was applied. The secondary system pressure behavior was also explicitly simulated by modeling the main steam isolation and steam relief valves with the imposed boundary condition for the feedwater flow following the reactor trip.

CCFL parameters were applied at SG U-tube inlet, hot leg, and crossover leg uphill side. The Hewitt and Wallis CCFL parameters were used for the SG U-tube inlet. The CCFL parameters for the hot leg were determined from the validation results of Dukler air-water flooding test analysis. The CCFL parameters for the crossover leg uphill were based on Tien [TEC15].

The new CHF multiplier for low flow and high void fraction conditions was also applied.

The pressurizer pressure decreased faster in the calculation than in the test because the Moody model overestimated the break flow in the two-phase regime. The M-RELAP5 prediction agreed reasonably well with the secondary pressure data.

Based on the analysis of the ROSA/LSTF SBLOCA (17 percent) LOCA test (IB-CL-02), the staff finds M-RELAP5 can adequately predict core dryout, post-CHF heat transfer, rewet, core mixture level, and downcomer mixture level phenomena occurring in the boiloff and core recovery phases in US-APWR SBLOCA. It was also confirmed that the US-APWR SBLOCA EM is conservative with respect to the predicted PCT.

4.4.2.4 LOFT SBLOCA (2.5 percent) test (L3-1)

LOFT L3-1 was included in Topical Report MUAP-07013-P in response to Request LS-1 [NRC08].

The LOFT L3-1 test was simulated with M-RELAP5 to assess the code ability to predict the US-APWR SBLOCA. The M-RELAP5 LOFT model was based on the input model developed by INL [DAT22]. However, the noding scheme and the thermal-hydraulic model options were modified to conform to the models applied to the US-APWR SBLOCA analysis.

The M-RELAP5 LOFT model primarily consisted of a) the RV, b) the pressurizer, c) the SG, d) the intact loop, e) the broken loop, f) the ECCS, and g) the break assembly. The RV was represented in the same manner as employed for the plant calculations. The pressurizer, SGs, hot leg and cold leg were nodalized using the same modeling approach used for the US-APWR noding. The ECC train including the accumulator, HPIS and Low Pressure Injection System (LPIS) were connected to the cold leg of the intact loop. The blowdown suppression vessel connected to the broken loop was represented with the time-dependent volume.

The heat conductors in the nuclear core fuel rods were divided axially into heat structures representative of the physical conductor geometry. Each heat structure was then radially subdivided into mesh intervals. In its response to Request LS-6 [MHI19], MHI confirmed the additional multiplier factor to the CHF was employed for the calculation.

HPIS, LPIS and accumulator injections were explicitly simulated. Although no significant effects from non-condensable gas from the accumulator were observed after the accumulator emptied, the non-condensable gas model simulating the nitrogen entering the RCS was used in this calculation.

The M-RELAP5 transient calculation simulated the experiment from the break initiation until shortly before the operators manually initiated the steam bleed of the secondary coolant system. The latter portion of the experiment was not simulated because the behavior of the LOFT facility, after the onset of the steam bleed, was not relevant to the behavior of the US-APWR.

The comparison of M-RELAP5 to the test data demonstrated that the model was capable of reproducing the experimental steady state.

The core fission power and decay power history were modeled with an input data table. The break flow was explicitly simulated with the Moody critical flow model with the offtake/pullthrough model, described in Section 4.2.4, “Break Pull-Through Model,” of this SE. The LOFT L3-1 test used a single-ended break unit with a centered orifice and a quick opening blowdown valve was attached in the downstream of the break orifice. This break configuration is different from that assumed for US-APWR plant calculations, and the geometry around the break unit was modeled to correctly simulate the break flow behavior. The post-test analysis report [DAT22] stated the steam control valve of the SG secondary system did not seat 100 percent nor did it seat the same for each closure. The value calculated for Experiment L3-1 was 0.02 kg/s (0.04 lbm/s) at 3.5 MPa (508 psia), and this value was used for the M-RELAP5 calculation.

The CCFL occurring in the piping with a smaller diameter was taken into account for the calculation. The CCFL in the SG U-tubes was modeled using the Wallis correlation [TEC13].

This modeling is identical to that for the US-APWR plant calculation because the geometric scaling of the SG U-tubes is almost identical between the LOFT and US-APWR.

For the hot leg piping, the Tien model [TEC15] was applied, whereas the US-APWR calculation employs the model developed from the UPTF experimental data which was obtained in the full-scale test facility.

Following break initiation, the RCS rapidly decreases to the secondary system pressure during the blowdown phase. The Moody critical flow model overestimated the two-phase break flow rate.

Around 1500 seconds after the break initiation, M-RELAP5 underestimated the break flow rate because the accumulator injection stops and the void fraction at the break location increased. The experimental test report noted that the uncertainty for the measured break flow rate was ±15 percent.

The temporal changes in the primary and secondary system pressures and the pressurizer liquid level were reproduced by M-RELAP5. The natural circulation phase began as the pumps complete their coast-down, and then the primary and secondary pressures decreased. Around 400 seconds after the break initiation, the primary system pressure fell below the secondary system pressure, which ended the natural circulation phase. After that, the SG no longer behaved as a heat sink.

Calculated differential pressures in the intact loop crossover leg downhill-side and uphill-side were compared with the measurements. The differential pressure was due to the liquid level in these regions after the natural circulation period ended. In the experiment, the loop seal that formed in the intact loop crossover leg was not cleared because the steam generated in the core was vented through the bypass paths. Reference DAT22 noted the initial core bypass fractions were 3.6 percent of primary loop flow for the lower plenum to upper plenum path, 6.6 percent for the inlet annulus (downcomer) to upper plenum path, and 1.3 percent for the reflood assist bypass valve at the test initiation. This large core bypass flow fraction prevented the coolant in the crossover leg from clearing. Similar to the measurement, the M-RELAP5 calculation predicted that the loop seal in the intact loop crossover leg did not clear throughout the transient.

The accumulator started injecting the safety coolant when the RCS pressure fell below the initial accumulator pressure. The accumulator emptied and the nitrogen began to enter the RCS. These behaviors were seen in the M-RELAP5 calculation, which validated the accumulator model implemented in M-RELAP5.

No fuel cladding heat-up was observed in the LOFT L3-1 test or calculated with M-RELAP5.

The LOFT L3-1 experiment was simulated with M-RELAP5 to validate the code’s ability to predict the plant response occurring under SBLOCAs. The primary purpose was to assess the M-RELAP5 models and noding scheme, which are also applied to the plant analysis, using the experimental test data.

M-RELAP5 showed reasonable agreement compared to the measured RCS pressure and the pressurizer, loop seal, and accumulator behaviors. Because of the large core bypass fraction, the loop seal in the crossover leg did not clear during the LOFT L3-1 test or in the M-RELAP5 calculation. M-RELAP5 predicted no cladding heat-up during the test, which was consistent

with measured results. MHI concluded that M-RELAP5 is able to reproduce the transient behavior, phenomena and processes of interest during the LOFT L3-1 SBLOCA test.

Based on the analysis of the LOFT SBLOCA (2.5 percent) test (L3-1), the staff finds M-RELAP5 can adequately predicts the US-APWR SBLOCA plant responses, including the RCS pressure and the pressurizer, loop seal, and accumulator behaviors.

4.4.2.5 Semiscale SBLOCA (5 percent) test (S-LH-1)

Semiscale S-LH-1 was included in Topical Report MUAP-07013-P in response to Request LS-1 [NRC08].

The facility was nodalized in the same manner as the US-APWR SBLOCA calculations. The M-RELAP5 Semiscale Mod-2C model consisted of a) the RV, b) the downcomer pipe, c) the pressurizer, d) the SG, e) the intact loop, f) the broken loop, and g) the ECCS.

The core was represented with one thermal-hydraulic channel, since the active core region consists of a small fuel assembly with a 5x5 heater rod array without radial power peaking. The heated region was nodalized similarly to the approach taken in the US-APWR. The core bypass path was explicitly modeled between the upper head and downcomer. Heat conduction in the electrical heater rod was modeled in each core hydraulic cell in the radial direction. The additional multiplier factor to the CHF was employed for the calculation. This model was also validated in the code assessment using the ROSA/LSTF test data.

The noding scheme for the downcomer, pressurizer, SG primary and secondary sides, crossover leg and cold leg were the same as used for the plant calculation. The hot leg, however, was nodalized with more cells to properly model the region between the first vertical bend and the SG inlet plenum. The geometry of this region differs from that of the US-APWR. The accumulator component and the time-dependent volume simulating the pumped SI were connected to the cold legs of both intact and broken loops. In modeling the Semiscale Mod-2C facility, the heater blankets surrounding the system were explicitly modeled and simulated because the heat loss from the system affected the transient behaviors. HPIS and accumulator injections were explicitly simulated in the model in the same manner as the plant calculation.

The comparison of M-RELAP5 to the test data showed that the model was capable of reproducing the experimental steady state conditions.

The core fission power and decay power history were defined by an input data table for the calculation. The break flow was explicitly simulated with the Moody critical flow model with the atmospheric boundary condition. Since the break unit was perpendicularly attached to the cold leg (a side orientation), offtake liquid entrainment and/or pullthrough vapor carryover effects were possible when horizontally stratified flow formed in the cold leg. The offtake/pullthrough model, described in Section 4.2.4, "Break Pull-Through Model," of this SE, was applied. The secondary system pressure behavior was also explicitly simulated by modeling the main steam isolation and steam relief valves with the imposed boundary condition for the feedwater flow following the reactor trip.

CCFL occurring in the vertical piping with a smaller diameter was taken into account for the calculation, specifically, the vertical part of the hot leg and SG U-tubes. This modeling was identical to that for the US-APWR plant calculation because the geometric scaling of the SG U-tubes was almost identical between the Semiscale and US-APWR. For the vertical portion of the

hot leg piping, the CCFL model developed by Kim and No [TEC17], which represents the CCFL phenomena as a function of the horizontal pipe L/D ratio, was applied. MHI described the Kim and No CCFL model in its response to Request LS-5 [MHI19].

The transient calculation was initiated by opening the break valve, and the simulated break flow was compared with the measurement. The Moody critical flow model overestimated the two-phase break flow; however, the result indicates that the model reasonably reproduces the measured data without excess conservatism.

The secondary system pressures for the intact and broken loops were also compared with the measurements. In the experiment, a steam leak occurred in the SG secondary system, and the secondary system pressure remained below the relief valve opening pressure setpoint even after the main steam isolation valve was closed. For the calculation, 0.5 percent was assumed for the steam leakage fraction in the calculation based on the Reference DAT23. The analysis indicated the steam leakage might be greater than the reported value. The overestimation in the secondary system pressure resulted in a slightly higher calculated pressure in the primary system during the natural circulation phase from about 50 to 170 seconds.

A complicated loop seal behavior was observed in the Semiscale S-LH-1 test, where the coolant seal in the intact loop cleared first and the broken loop seal cleared later. These loop seals were simulated in M-RELAP5 and the code's ability to predict the loop seal behavior during SBLOCAs was confirmed. MHI noted that M-RELAP5 predicted a transient decrease in the collapsed liquid level for the broken loop crossover leg since the core liquid level was depressed during the loop seal period, while this was not observed in the measurement. However, the resultant core liquid level depression predicted by M-RELAP5 was deeper than the measurement, resulting in a conservative prediction with respect to the loop seal PCT.

Significant reflux flooding occurred in the hot leg piping and SG U-tubes in the S-LH-1 test, and the core liquid level was significantly depressed during the loop seal phase. This was principally caused by the small core bypass flow fraction between the upper head and downcomer, which prevented steam venting from the core. This was experimentally validated by comparing two tests, S-LH-1 (0.9 percent bypass) and S-LH-2 (3.0 percent bypass), from the Semiscale Program [DAT24]. The M-RELAP5 results were compared with the measurements for the hot leg and SG U-tubes, for the intact and broken loops, and the core liquid level. The reflux flooding and core liquid depression were adequately simulated with M-RELAP5. As a result of the core liquid depression, the heater rod experienced dryout and heat-up during the loop seal phase. This temperature excursion was terminated by an increase in the core liquid level after the loop seal cleared. The difference between the calculated and measured cladding temperatures prior to scram was shown by the comparison of the calculated surface temperature with the measured temperature inside the heater rod.

Since M-RELAP5 predicted a deeper core liquid level depression, the highest heater rod cladding temperature was obtained at the 5.7-ft elevation, while the highest temperature occurred at the 8.3-ft elevation in the experiment.

M-RELAP5 used a chopped-cosine axial power shape to represent the heater rod, and, therefore, M-RELAP5 showed a higher peak cladding temperature than measured during the loop seal clearing phase. M-RELAP5 showed a conservative prediction for the heater rod temperature, even for the elevation where the highest experimental temperature was measured. In the experiment, the heater rod heat-up was terminated when the seal cleared in the intact loop. In the calculation, the heat-up was terminated when the intact loop seal cleared, but

started again at a higher elevation due to a larger liquid level depression in the core. The second heat-up was terminated around 250 seconds, after the broken loop seal cleared.

MHI concluded the results showed that M-RELAP5 predicted the complicated plant responses, including the significant fluid holdup in the hot leg and in the SG U-tubes, and the loop seal behaviors. In particular, the core depression and heater rod temperature excursion during the loop seal phase were reproduced by M-RELAP5, resulting in a conservative prediction for the heater rod cladding surface temperature.

Based on the analysis of the Semiscale SBLOCA (5 percent) test (S-LH-1), the staff finds M-RELAP5 can adequately predict the US-APWR SBLOCA plant responses, including the significant fluid holdup in the hot leg and in the SG U-tubes, and the loop seal behavior. This analysis also demonstrates that the US-APWR SBLOCA M-RELAP5 EM predicts conservative cladding surface temperature.

4.5 Scaling

MHI performed a top-down and a bottom-up scaling analysis for each of the five SBLOCA accident phases. The 7.5-inch and 1.0-ft^2 breaks were discussed separately. The top down scaling analyses utilized ROSA-IV/LSTF test SB-CL-18 for the 7.5-inch break and test IB-CL-02 for the 1.0-ft^2 break. Reduced governing conservation equations were developed by MHI for each phase of the event. The equations were non-dimensionalized and applied for the test facility and for the US-APWR plant, following the method developed for the AP600 SBLOCA and documented in INEL-96/0040 [TEC06]. The same equations were applied for both break sizes. The reference conditions used to evaluate the dimensionless groups were selected as appropriate for each break size. The non-dimensional coefficients in the simplified governing equations characterized the system response. Large differences were indicative of potential distortions in the test facility compared to the plant for the particular phase of the event being analyzed. When any significant scaling distortions were identified between the US-APWR and ROSA-IV/LSTF, the effect was evaluated based on the bottom-up scaling approach.

The reduced equations were solved numerically. Applicability of the developed reduced models was verified by comparing the normalized responses of the key variables predicted by the reduced model with results from M-RELAP5 calculations and experimental measurements from the ROSA/LSTF SB-CL-18 and IB-CL-02 tests.

The scalability acceptance criterion was defined as the test facility and prototype (the US-APWR) that have the same governing equations and the relative importance of the dominant coefficients is also the same. Since the US-APWR and ROSA share the same configuration, the staff finds the scalability criterion acceptable to show that the same behavior can be expected in both systems. Bottom up scaling results presented by the applicant further confirmed applicability of the test data from the ROSA-IV/LSTF tests as acceptable for validating the M-RELAP5 code for application to the US-APWR SBLOCA event.

4.5.1 Blowdown Phase

The blowdown phase starts at the break initiation and ends when the primary system pressure nearly equals the secondary pressure. The RCS depressurization initiated by the break is a dominant global behavior during the blowdown phase. The discharge flow out the break determines the initial drop of RCS inventory, which affects the depressurization rate and the

duration of blowdown. In the US-APWR system, transient behavior of the pressurizer pressure determines the timing for the reactor trip (scram) and the SI. Therefore, the primary system mass and depressurization were addressed as significant parameters of interest for the blowdown phase.

Top-Down Scaling Analysis

The applicant modeled the system in this phase with one control volume with two connected regions, representing the subcooled regions in the system and the two-phase regions in the system, and the break. Mass conservation and energy conservation equations were developed for this system, and the energy equation was expressed as a dynamic pressure equation, as in Idaho National Engineering Laboratory [INEL]-96/0040 [TEC06]. A comparison of the non-dimensional coefficients in the reduced equations for the ROSA-IV/LSTF facility and the plant was presented in Table 6.1-2 of the scaling report for the 7.5-inch break and in Table 6.1-4 for the 1.0-ft^2 break. The ratios of the non-dimensional coefficients reasonably agree, except for the ratio of pressure change due to change in specific volume of the subcooled region from volumetric flow, to reference pressure for the 7.5-inch break. The value for this ratio (pressure change due to change in specific volume) indicated a distortion in the volumetric flow to the break.

Solution of the reduced equations showed that the pressure and system mass response for the test facility and the plant were comparable. Validity of the solution of the reduced equations was established by comparing the pressure and mass responses in the ROSA-IV/LSTF data and the M-RELAP5 calculations for the complete plant model. The comparisons with M-RELAP5 calculations, shown in Figure 6.1-22 and Figure 6.1-23 of the scaling report [MHI17] with the experimental data in Figure 6.1-26 and Figure 6.1-27 confirmed that the reduced equations provide an acceptable representation of the plant and the test facility for the purpose of demonstrating that the experimental data is appropriate for use in validating the M-RELAP5 code for the blowdown portion of the event.

The applicant discussed the distortion in ROSA break flow behavior in the first few seconds of the transient and its consequences on the timing of events afterward, the concern being the potential impact on departure from nucleate boiling (DNB). However, DNB is not expected to occur in this phase of the SBLOCA. In summary, acceptable comparisons of energy and mass non-dimensional variables in non-dimensional space and time showed that the two systems follow the same trends very closely, indicating that their behavior is quite similar. A distortion in the depressurization rate early in the blowdown was identified and evaluated to show that it did not affect the suitability of the experimental data for code validation purposes.

Bottom-Up Scaling Analysis

The important phenomena in this phase are primary side and secondary side heat transfer, critical flow and break flow enthalpy. The SG heat transfer did not exhibit distortions during this phase and the code calculations follow the experimental measurements closely. The US-APWR SBLOCA analyses use the Moody critical flow rate model to address critical flow.

Blowdown Phase Summary

The staff determined that the scaling analysis during the blowdown period demonstrates that the ROSA-IV/LSTF data used by the applicant are acceptable for validation of the M-RELAP5 code for the US-APWR SBLOCA blowdown phase. As use of the Moody critical flow model is

required by Appendix K, no bottom up scaling analysis was needed for this phenomenon even though there were some differences in predictions between the test facility and the US-APWR plant calculations. Implementation of the Moody critical flow model is addressed in Section 4.6.5 of this SE. The applicant adequately explained the difference in depressurization rate for the 7.5-inch break as due to the uncertainty in break flow.

4.5.2 Natural Circulation Phase

The natural circulation phase begins when the pumps have coasted down and ends when there is no substantial net liquid flow at the top of the SG tubes. The system, in this case, was a closed loop with heat added to the core and removed by the SGs, with mass being lost from the cold side of the loop through the break. The recirculating flow loop was broken when the liquid flow at the top of the U tubes ended. In addition to the break flow and the SG heat transfer, the PIRT identifies the void distribution in the lower plenum and the downcomer as important phenomena. The variables of interest during this phase are the depressurization rate and RCS mass inventory.

The above discussion applied for the 7.5-inch break. For the 1-ft^2break, there is no natural circulation or loop seal clearance phase following the blowdown, since the larger breakflow results in a rapid depressurization. The accumulator starts injecting safety coolant prior to the high-head injection system (HHIS). The accumulator flow rate exceeds the break flowrate, so the RCS mass inventory recovery starts. Since there was not a well-defined natural circulation phase, the applicant defined the boiloff time-period for the 1-ft^2break from the end of the subcooled discharge, to the time when the core liquid level becomes minimum. Therefore, subsequent discussions of the natural circulation and loop seal clearance phases apply only to the 7.5 inch break.

Top-Down Scaling Analysis

MHI modeled the primary system as a single control volume with heat transfer from the core and SGs and mass flow out of the break. Mass conservation and energy conservation equations were developed for this system, and the energy equation was expressed as a dynamic pressure equation, as in the referenced work. The applicant used a spreadsheet to evaluate the non-dimensional groups of Ψ5, Ψ6, Ψ11 and Ψ13 in the reduced non-dimensional equations for both US-APWR and ROSA-IV/LSTF.Ψ5 is the ratio of the pressure change, due to the change in specific energy of the saturated field from mass outflows, to the reference pressure. Ψ6 is the ratio of the pressure change, due to change in specific energy of the saturated field from heat transfer, to the reference pressure. Ψ11 is the ratio of the pressure change, due to change in specific volume of the saturated field from volumetric flow, to reference pressure. Ψ13 is defined as the ratio of net mass flow to the reference system mass.

Evaluation of the non-dimensional groups by the applicant revealed that the SG heat transfer in ROSA-IV/LSTF was disproportionately larger than for the US-APWR, likely due to the larger difference between primary and secondary pressures in ROSA/LSTF. This larger heat loss explained why the pressure was slowly decreasing in ROSA-IV/LSTF, while it was nearly constant in the US-APWR.
Evaluation of the non-dimensional groups also revealed that both the US-APWR and ROSA-IV/LSTF lose about the same amount of mass inventory during this natural circulation period, and the RCS mass inventory response of the ROSA-IV/LSTF facility is representative of the US-APWR RCS mass inventory response.

The top-down scaling of the system momentum balance was based on the methodology developed by M. Ishii and I. Kataoka for the two-phase natural circulation system [TEC18], and on the methodology applied to the AP600 scaling analysis by J. N. Reyes, Jr. and L. Hochreiter [TEC19]. The momentum conservation was considered independently from the mass and energy conservations. This follows the approach of Reyes and Hochreiter. Each of the scaling ratios of non-dimensional numbers indicated that the ROSA-IV/LSTF natural circulation was acceptably scaled from the momentum point of view. For the single-phase region, there appeared slight scaling distortions in the friction and orifice numbers.

However, the integral of the friction and orifice numbers provided an acceptable scaling ratio between ROSA-IV/LSTF and the US-APWR.

As in the blowdown phase, applicability of the reduced models was verified by comparing the normalized responses of the key variables predicted by the reduced model with results from M-RELAP5 calculations and experimental measurements. In both the US-APWR and ROSA/LSTF, the natural circulation phase terminated when the RCS lost about the same amount of the initial inventory. Both M-RELAP5 and the reduced model calculated a consistent non-dimensional depressurization rate for the US-APWR 7.5-inch cold-leg break. The calculation and the application of the reduced model both demonstrate for the US-APWR, that the natural circulation phase in a SBLOCA occurs at a nearly constant pressure.

These results demonstrated that the ROSA-IV/LSTF test data are suitable for validating the M-RELAP5 code to calculate total system mass inventory at the end of the natural circulation phase of a SBLOCA in the US-APWR.

Bottom-Up Scaling Analysis

MHI examined three local aspects of the system behavior: steam generation in the core, two-phase flow in the piping, and time scale in the piping.

The steam generation in the core was examined by evaluating and comparing the “phase change number” as proposed by Ishii and Kataoka. While the magnitude of this parameter was different for ROSA-IV/LSTF and US-APWR during the progression of this phase, the trends were similar, and the ROSA-IV/LSTF behavior contained that was expected in US-APWR, demonstrating the adequacy of the data for code validation of this phenomenon.

For the two-phase flow in the hot legs, MHI evaluated the Froude number as suggested by Zuber [TEC20]. The progress of this non-dimensional parameter through this phase shows that the ROSA-IV/LSTF behavior contained that expected in the US-APWR, again demonstrating the adequacy of these data to validate this phenomenon.

MHI investigated the time scales of the facility’s piping, using Zuber’s suggested ratios, and concluded that ROSA-IV/LSTF is about 11 percent faster than the US-APWR. The staff concluded this small piping time scale difference did not have an impact on the results obtained from the scaling analysis.

Natural Circulation Phase Summary

Since the non-dimensional groups for the ROSA-IV/LSTF facility and the plant were of the same relative size, the experimental data are acceptable for validation of the M-RELAP5 code for simulation of SBLOCA events in the US-APWR. The applicant examined local aspects of the

system behavior: including steam generation in the core, two-phase flow in the piping, and time scale in the piping. The staff accepts the evaluation of the local aspects of the important phenomena performed by the applicant to establish that they have no impact on the acceptability of the data for M-RELAP5 code validation.

4.5.3 Loop Seal Clearance Phase

The loop seal phase begins on termination of natural circulation and ends when the loop seal(s) clears. When natural circulation ends, the top regions of the system become pockets of steam, while the low regions contain liquid. By virtue of its location in the system, the break energy outflow is restricted by its low quality. During this phase, the core may experience uncovery and a temperature excursion. When the liquid in the loop seal is cleared, steam can come out the break increasing the depressurization rate and allow SI to replenish the inventory in the vessel. The core liquid level is a significant parameter of interest for this phase because of its potential impact on the core dryout.

Top-Down Scaling Analysis

Non-dimensionalizing the governing mass conservation and pressure equations and evaluating the non-dimensional coefficients showed that the process hierarchy was preserved from one facility to the other, thus assuring that physical processes occurring during this phase are comparable.

The coefficients in the level equation suggested that ROSA-IV/LSTF will show a lower core level than US-APWR, resulting in a minor distortion. The deeper loop seal in ROSA-IV/LSTF resulted in a slightly lower core liquid level. This lower core level response makes the ROSA-IV/LSTF conservative relative to the US-APWR. The applicant provided plots of the normalized pressure (Figures 6.3-17 and -19) and inventory (Figure 6.3-16 and -18) for this phase [MHI17]. The experimental trace for ROSA-IV/LSTF and the calculated trace for US-APWR merged together for the entire span of this phase. Therefore, the important mass inventory response was similar for the two systems. No significant scale distortion was observed in the mass and pressure equations during the loop seal clearance phase.

Bottom-Up Scaling Analysis

In the bottom-up analysis, the applicant examined CCFL in the hot leg, CCFL in the SG-U tubes, and water retention in the crossover leg.

For the CCFL in the hot leg, the applicant verified that the ROSA-IV/LSTF data spanned the range expected in US-APWR when predicted using the Kutateladze number with empirical CCFL values from UPTF data. These values were based on UPTF data which also covered most of US-APWR's expected range. The difference in pressure between UPTF and US-APWR was evaluated by MHI, which concluded the US-APWR calculation was conservative.

For CCFL in the SG U-tubes, the Wallis correlation was used. Applicability of the data was established using the Dukler air-water flooding test. Figure 6.3-23 of the scaling report compared J* between ROSA-IV/LSTF and US-APWR during this phase, showing significant overlap.

Similarly, to assess the adequacy of the data for water retention in the crossover leg, MHI showed an acceptable comparison of M-RELAP5 predictions and UPTF Test 5 data (Figure 6.3-

24 in the scaling report [MHI17]). In Figures 6.3-25, the applicant compared the non-dimensionalized pressure difference in the upflow side of the loop seal for ROSA/LSTF and US-APWR, and showed that the traces followed each other.

The temporal variation of differential pressure along the upflow-side of loop seal between the US-APWR and the ROSA-IV/LSTF was compared in Figure 6.3-25. The differential pressure was normalized to the initial value when the differential pressure started to decrease.

The time was also normalized to the period during the differential pressure decrease. The temporal variation was similar between the two systems even with the difference in pipe diameters, justifying the use of ROSA-IV/LSTF data for M-RELAP5 code validation of loop seal water retention.

Loop Seal Clearance Phase Summary

Core liquid level is primarily controlled by the CCFL induced liquid head in the uphill side of SG U-tubes and inlet plena, and by the head balance caused by the distribution of liquid along the loop seal. The top down scaling analysis established that the process hierarchy was preserved from one facility to the other, thus assuring that physical processes occurring during this phase are comparable for ROSA-IV/LSTF and the US-APWR. However, the core liquid level is likely to be more depressed in ROSA-IV/LSTF compared to the US-APWR due to the deeper loop seal in ROSA-IV/LSTF. The adequacy of loop seal behavior predicted for the US-APWR was also confirmed by the assessment for the residual water prediction in UPTF tests performed as a portion of the bottom-up scaling analysis. For these reasons, the staff finds the ROSA-IV/LSTF and SET data used by the applicant acceptable for validating the M-RELAP5 code for determining the US-APWR system response during the loop seal clearing phase of a SBLOCA.

4.5.4 Boiloff Phase

The boiloff phase commences at the end of the loop seal clearance phase and continues until the RCS mass inventory starts recovering. During this phase, mass inventory continues to be lost through the break so that core liquid level can decrease leading to heatup of the fuel. The RCS liquid inventory and the processes that control it (depressurization, boiling, break flow) are the important phenomena. Based on the PIRT, the important phenomena and processes during the boiloff phase are the CHF/dryout, uncovered core heat transfer, and the mixture level in the core and RV.

There was an apparent scaling distortion of the ROSA-IV/LSTF facility because there was no pumped SI in the test. This was shown in the comparison of the normalized RCS mass response between the US-APWR 7.5-in break and ROSA test SB-CL-18, in Figure 6.4-20 in the scaling report [MHI17].

Top-Down Scaling Analysis

The governing mass and energy conservation equations for a two-phase mixture in a boiler tank were non-dimensionalized. As with earlier phases of the event, acceptability of the reduced model was verified to ensure the reliability of the results. The normalized mass and pressure responses reproduced by the reduced model were compared with results from the M-RELAP5 calculations (US-APWR SBLOCAs) and experimental measurements (ROSA/LSTFSB-CL-18 and IB-CL-02), in Figure 6.4-24 through Figure 6.4-27 in the scaling report [MHI17]. These

comparisons demonstrated that the reduced model reproduces the code calculated boiloff responses. Therefore, the staff concluded that the top-down scaling results are acceptable.

The coefficients in the non-dimensional equations are dependent on the mass and pressure at the beginning of the phase and the relative values of mass and energy inputs and outputs. The coefficients for both systems were close to unity, indicating similarity between the experimental facility and the plant. The scaling report provided numerous figures to support this. In addition, MHI showed that the test was conservative with respect to the plant.

Bottom-Up Scaling Analysis

Considering no HHIS operation in the ROSA-IV/LSTF test, the differences in the RCS mass responses were explained, and were not a critical scaling concern. The lack of pumped ECC flow in ROSA-IV/LSTF makes the ROSA results conservative relative to the US-APWR.

The applicant used data from the ORNL/THTF test facility to validate the CHF/Dryout model in M-RELAP5. The same data were used to validate uncovered core heat transfer. To validate the two-phase mixture level predictions, the applicant used the measured void profiles in ROSA/LSTF tests and ORNL/THTF's tests. The scaling report [MHI17] contains several figures (Figure 6.4-32, Figure 6.4-33, and Figure 6.4-34) which showed that the data range contained the conditions expected in the US-APWR.

BoilOff Phase Summary

The RCS mass and pressure responses during the boiloff phase are important because they determine the core liquid level depression, the pumped SI flow rate, and the accumulator actuation, which affect the core heat-up behavior. Similarity of the global RCS mass and pressure responses were investigated for the US-APWR SBLOCA and ROSA-IV/LSTF tests SB-CL-18 and IB-CL-02. The top-down scaling analysis demonstrated that the ROSA/LSTF SB-CL-18 test is well-scaled to the US-APWR 7.5-inch Cold Leg Break. The dominant nondimensional coefficients were the same between the two systems and the order of ranking of nondimensional coefficients was similar. This means that the global processes observed in the US-APWR SBLOCA and ROSA test were the same.

The CHF/dryout, uncovered heat transfer, and two-phase mixture level were identified by the applicant as important phenomena and processes during the boiloff phase. These phenomena and processes were validated for M-RELAP5 analysis of US-APWR using test data obtained in the ORNL/THTF and ROSA-IV/LSTF test facilities. The experimental test conditions, pressure, temperature, flow rate, and power were compared with those expected for a US-APWR SBLOCA. The staff determined that US-APWR SBLOCA conditions are covered by the selected experimental tests.

For the reasons discussed above the staff finds that the experimental data from the ROSA-IV/LSTF and the referenced SETs acceptable for validating the M-RELAP5 code for calculating US-APWR system response during the boiloff phase of a SBLOCA.

4.5.5 Core Recovery Phase

The core recovery phase starts when SI mass inflow exceeds the break flow and the RCS inventory begins to increase, and ends when core recovery is completed. RCS mass inventory

is the primary response of interest in the top-down scaling. Core reflooding and rewetting are important local thermal-hydraulic behaviors considered in the bottom-up scaling analysis.

The RCS pressure during the core recovery phase was lower in the 1-ft^2break than in the 7.5-inch break and in the ROSA-IV/LSTF SB-CL-18. The difference in pressure response was evaluated in the applicant's bottom-up scaling analysis.

Top-Down Scaling Analysis

The reduced system equation in this phase consisted of a simple volume since the only response of concern was related to the system mass. This yielded a single non-dimensional coefficient defined as the ratio of net mass flow to the reference system mass. The US-APWR 7.5-inch break with ROSA-IV/LSTF test SB-CL-18 coefficients had essentially identical values. The comparison of the dimensionless group for the 1 ft^2 break and ROSA-IV/LSTF test IB-CL-02 was within the range that demonstrated similarity in the physical phenomena.

The normalized RCS mass response from the reduced model was compared with US-APWR M-RELAP5 calculations for the SBLOCA and with measured results for the ROSA-IV/LSTF SB-CL-18 and IB-CL-02 tests to validate the reduced model. The applicant demonstrated that the reduced model was capable of reproducing the mass inventory response for both the US-APWR 1-ft^2 break and the ROSA-IV/LSTF IB-CL-02 test and the US-APWR 7.5-inch break and ROSA-IV/LSTF test SB-CL-18.

Bottom-Up Scaling Analysis

For the 1-ft^2 break case, a larger ratio of mass inflow to the initial system mass was seen for the US-APWR compared to the ROSA-IV/LSTF. This was mainly caused by a lower initial system mass when the core recovery phase started due to the use of the Moody Appendix K break flow model In the US-APWR. The Moody break flow model tends to overestimate the break flow rate for this break size resulting in the smaller initial system mass and lower system pressure when the core recovery phase started. The explanation of the distortion was consistent with the difference in system responses. Based on this explanation of the distortion and the acceptable results of the top-down scaling analysis and bottom-up scaling, the ROSA-IV/LSTF data are acceptable for validating the M-RELAP5 code during the recovery phase of a US-APWR SBLOCA event.

Experimental data from the ORNL/THTF high-pressure reflood tests and FLECHT-SEASET forced-reflood tests were used to validate the reflooding processes and rewet phenomena. The staff determined that the primary dimensions of the SET facilities employed by the applicant were well scaled to the US-APWR design, and the experimental test conditions adequately cover the range of conditions expected for the US-APWR SBLOCAs.

Core Recovery Phase Summary

The top-down scaling analysis demonstrated that the ROSA-IV/LSTF SB-CL-18 and IB-CL-02 tests can be used to validate the M-RELAP5 code with respect to the RCS mass response of the US-APWR SBLOCAs because of the favorable comparison of the dimensionless coefficients in the reduced governing equations. The ROSA-IV/LSTF test SB-CL-18 was well scaled to the 7.5 inch break in the US-APWR. For the US-APWR 1-ft^2 break the Appendix K required Moody break flow model used in M-RELAP5 resulted in a smaller mass inventory and lower system pressure at the start of the core recovery phase of the SBLOCA. However, this

distortion was adequately explained by the applicant. Therefore, based on the acceptable results from the top down scaling analysis, the applicant's explanation of the distortion and results of the bottom-up scaling analysis, the staff finds the ROSA-IV/LSTF data acceptable for validating the M-RELAP5 code during the core recovery phase of US-APWR SBLOCAs.

The bottom-up scaling analysis showed that the primary dimensions of the ORNL/THTF high-pressure reflooding test and the FLECHT-SEASET forced-reflood test facilities were well scaled to the US-APWR design and the experimental test conditions adequately covered the range of conditions expected for the US-APWR SBLOCAs.

4.6 Appendix K Requirements

4.6.1 Gap Conductance Model

The Appendix K item I.A.1 requirement related to the fuel-to-cladding gap heat transfer, is that steady state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature. The fuel-to-cladding gap heat transfer, or the gap conductance, is used to calculate the initial stored energy and transient heat transfer across the gap. For the US-APWR SBLOCA analysis, the initial fuel temperature is adjusted to that calculated by the MHI fuel design code, FINE [MHI34], which is a detailed fuel rod design computer code that considers burn-up effects on the fuel temperature.

The FINE gap heat transfer model is based on the pellet concentric annular gap model. The pellet concentric annular gap model was implemented in M-RELAP5 to maintain consistency with the FINE fuel design code. RELAP5-3D also accounts for heat transfer by thermal radiation across the gap. This term was retained in M-RELAP5.

The staff, in RAI 7-3 [NRC03], requested that MHI verify the implementation of the gap heat transfer model and the radiation conductance model across the gap in M-RELAP5. MHI provided a verification analysis in [MHI03]. A simple M-RELAP5 model was used to perform the study, consisting of a single fuel rod in a single fluid volume, with the inlet and outlet conditions set to make the M-RELAP5 cladding surface temperature agree with the FINE value and to establish the fluid volume pressure. The internal rod pressure calculated by FINE was used in M-RELAP5. The input values for the surface roughness of the fuel and cladding for use in M-RELAP5 were selected so that the effect of the surface roughness on the gap conductance was the same for both M-RELAP5 and FINE.

The M-RELAP5 calculated gap conductance using the pellet concentric gap conductance model, showed a better agreement with the FINE prediction than the RELAP5-3D pellet offset gap conductance model.

Therefore, MHI selected the pellet concentric gap conductance model for use in the US-APWR SBLOCA analyses. However, there was a small difference between the M-RELAP5 and FINE predictions. MHI attributed this to the different thermal expansion models for the cladding and the fuel in M-RELAP5 and in FINE. These differences affected the predicted gap size in each code. When the gap conductance calculated with M-RELAP5 was divided by the ratio of the gap size calculated with FINE to the gap size calculated with M-RELAP5, the scaled M-RELAP5 calculated gap conductance agreed well with that from FINE. MHI concluded that the pellet concentric gap conductance model, consistent with FINE model, was satisfactorily implemented

in M-RELAP5. The staff accepts the pellet concentric gap conductance model based on this verification study.

MHI also compared the thermal radiation portion of the overall gap conduction with the RELAP5-3D thermal radiation model, to verify the code changes did not impact the radiation model. The M-RELAP5 predicted radiation conductance agreed with the RELAP5-3D model.

MHI concluded that the radiation conductance model across the gap was satisfactorily implemented in M-RELAP5. The staff finds that the radiation conductance model has been verified to be correctly implemented in M-RELAP5.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.A.1, “Initial stored energy in fuel,” for use in the evaluation of the US-APWR ECCS performance for SBLOCA analyses.

4.6.2 Fission Product Decay

The Appendix K item I.A.4 requirement specifies that the 1971 ANS decay heat model standard [TEC21] for the evaluation of the decay heat from fission products, multiplied by 1.2, should be used assuming the reactor has been operating at a constant total power for an infinite period of time. The 1973 ANS standard [TEC22] was modeled in RELAP5-3D. The existing RELAP5-3D decay heat model was modified by MHI to reproduce the 1971 ANS standard model when evaluating the decay heat for an SBLOCA event. A revised set of coefficients, based on the 1973 model, was developed by MHI for M-RELAP5. A comparison of the M-RELAP5 decay heat prediction to the 1971 standard showed good agreement. M-RELAP5 slightly overpredicts the decay heat when compared to the 1971 standard, except for a small period of time around 10 seconds following reactor shutdown. MHI concluded that the revised set of coefficients for use with the 1973 standard satisfies the requirement for the use of the 1971 standard, with M-RELAP5 conservatively predicting the decay heat. The staff agrees that the revised set of coefficients for use with the 1973 standard satisfies the requirement to use the 1971 standard.

The Appendix K item I.A.3 requirement specifies that the decay heat from actinides shall be calculated in accordance with fuel cycle calculations and known radioactive properties and shall be appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA. The point kinetics model in RELAP5-3D includes the ANSI/ANS 5.1-1979 [TEC23] standard decay heat model for the actinide series. This model was accepted by the NRC in Section 9.2.5, “Ultimate Heat Sink,” (Revision 3) in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” [NRC11] and is acceptable to the staff.

The staff, in RAI 7-4 [NRC03], requested that MHI provide a reference, or fuel cycle calculation, to justify its position that the default values of the ANSI/ANS 5.1-1979 standard are appropriate for the US-APWR and yield the highest decay heat from the actinide series. MHI provided an assessment in Reference MHI03.

The default values include the yield of ^{239}U produced per nuclear fission, the released energy from the decay of an actinide nucleus and the decay constant. Since the released energy from the decay of an actinide nucleus and the decay constant are specific values for each actinide nucleus, they are independent from the fuel cycle calculations for the US-APWR; MHI concluded the default values are appropriate for the US-APWR. However, the ^{239}U yield per nuclear fission is dependent on the fuel specifications, in particular the fuel enrichment and the

fuel burnup. Calculated results of the ^{239}U yield for the US-APWR, based on the expected fuel enrichment, were bounded by the default yield value of 1.0 in M-RELAP5 over a broad range of fuel burnup, up to about 75,000 MWD/MTU. Therefore, MHI concluded it is conservative to use the default value 1.0. The staff finds the use of the default yield value acceptable.

The staff concludes that M-RELAP5 satisfies the Appendix K requirement for I.A.4, "Fission product decay heat," and I.A.3, "Actinide decay heat," for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.3 Metal Water Reaction Model

The Appendix K item I.A.5 requirement is: "The rate of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just equation." The metal-water reaction model in RELAP5-3D, the Cathcart model [TEC24], did not meet this Appendix K requirement. The Baker-Just equation [TEC25] was incorporated into M-RELAP5 by MHI. The Baker-Just equation will bound the estimated oxidation for the advanced zirconium alloy cladding material, ZIRLO™, used in the US-APWR.

The staff, in RAI 7-5 and RAI 7-6 [NRC03], requested MHI verify the implementation of the Baker-Just equation into M-RELAP5. MHI provided a verification analysis in Reference MHI03. A simple M-RELAP5 noding scheme representing a fuel rod with a 100 mm (3.94 in) axial length and a single fluid volume representing the fluid volume containing the fuel rod were used. A time- dependent volume maintained the cladding surface temperature at 1200 ºC (2192 ºF). Fluid pressure was maintained at 10 MPa (1450 psia), using a time-dependent volume. Fission and decay heat power were neglected and a large rod surface heat transfer coefficient was used so that the metal-water reaction heat rate could be estimated from the heat transfer rate from the cladding to the vapor. Cladding swelling and rupture calculations were bypassed so that the M-RELAP5 and analytical results could be readily compared. The comparisons of the calculated results of the zirconium reacted thickness, the reaction heat release rate and the hydrogen mass generated from M-RELAP5 with the analytical (Baker-Just equation) results showed M-RELAP5 agreed with the analytical method. MHI concluded that the Baker-Just correlation was satisfactorily implemented into M-RELAP5 and the hydrogen generation rate and the heat generation rate were satisfactorily calculated. Based on this verification study, the staff concludes MHI has satisfactorily implemented the Baker-Just equation into M-RELAP5.

The staff concludes that M-RELAP5 satisfies the Appendix K requirement for I.A.5, "Metal-water reaction rate," for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.4 Cladding Swelling and Rupture Model

The Appendix K item I.B requirement is: "Cladding swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables."

The cladding swelling and rupture behavior of the US-APWR advanced zirconium alloy cladding material, ZIRLO™, are different from those of Zircaloy-4 cladding because of differences in the phase change temperatures. New cladding swelling and rupture models for ZIRLO™ have been developed from a series of test programs [TEC26]. These new models were incorporated into M-RELAP5 by MHI for the cladding rupture, the cladding strain at rupture, and the flow

blockage for ZIRLO^{TM}. The staff finds these new models acceptable because they are based on testing for the ZIRLO^{TM} cladding.

In RELAP5-3D, the cladding plastic hoop strain before rupture is calculated using the FRAP-T6 [TEC27] high temperature creep model. This correlation was developed for Zircaloy-4 cladding and is also used in M-RELAP5, however, the ZIRLO^{TM} cladding material properties are used to calculate the plastic hoop strain. The staff finds the use of this model with ZIRLO^{TM} material properties acceptable.

When the rod ruptures, additional form loss coefficients, which cause flow diversion from the cladding rupture region, are applied to the junctions just below and just above the rupture location. Experimental studies have shown that the effect of the flow diversion can be offset by a heat transfer enhancement from the flow blockage. However, MHI does not take any credit for the heat transfer enhancement due to flow blockage, which the staff agrees is a conservative approach.

The temperature through a cylindrical heat structure is calculated at fixed mesh points with the heat conduction equation in RELAP5-3D. If the cladding geometry is changed because of plastic hoop strain or rupture, the effect on the heat conduction calculation needs to be taken into account. A methodology to account for the effect of the cladding geometry change was implemented in M-RELAP5 by MHI, as described in Appendix B of Topical Report MUAP-07013-P.

In the MHI model, when cladding plastic strain begins or cladding rupture occurs, new temperature calculation mesh points are determined so that the volume represented by each calculation mesh point is preserved. Using this approach, the cladding temperature after deformation can be evaluated using the calculation mesh points before cladding deformation when appropriate corrections to the cladding thermal conductivity, gap heat transfer coefficient and the wall heat transfer coefficient are used. These corrected values are determined by a multiplier based on the ratio of the new mesh point radius to the old mesh point radius. With this model, the cladding temperature after deformation can be evaluated without changing the calculation mesh points used with the heat conduction equations in M-RELAP5.

MHI considers the cladding deformation from thermal expansion or elastic deformation, resulting from the pressure difference between the inner and outer surfaces of the cladding, to have little effect on the temperature distribution in the fuel. Therefore, these deformation effects are neglected in the heat conduction calculations in M-RELAP5, consistent with the original RELAP5-3D model. The staff finds the model to account for the effect of a cladding geometry change on the heat conduction acceptable.

The other part of this Appendix K requirement, related to the fuel-cladding gap heat transfer, is that "…the gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables." The fuel-cladding gap dimension, at any time, is calculated considering the pellet/cladding thermal expansion and cladding elastic deformation and the cladding swelling and rupture to obtain the gap conductance; therefore, this requirement is satisfied. MHI has implemented ZIRLO^{TM} specific models in M-RELAP5.

After cladding rupture is calculated, the thermal conductivity of steam is used as the fuel gap conductance at the rupture node. Since the gap width is usually large at the rupture node, the effects on the temperature jump distance and fuel/cladding surface roughness is expected to be

small. Therefore, MHI neglects these effects for the rupture node in M-RELAP5. The staff agrees these effects are small and agrees that they can be neglected.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.B, “Cladding swelling and rupture,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.5 Discharge Model

The Appendix K item I.C.1b requirement specifies the Moody critical flow model [TEC28] shall be applied for the evaluation of the break discharge when the conditions at the break location are two-phase. MHI incorporated the Moody critical flow model into M-RELAP5 for the evaluation of the break discharge flow for the US-APWR SBLOCA licensing analyses. In the Moody model calculation, the values of the upstream cell center are used to define the stagnant pressure, enthalpy and entropy for use in evaluating the critical flow rate. The selection of the critical flow model used to determine the break flow rate is based on the equilibrium quality.

The staff, in RAI 7-9 [NRC03], requested that MHI address the operation of the US-APWR advanced accumulator, which might introduce noncondensable gas into the system and which would require use of the extended Henry-Fauske model [TEC29]. MHI addressed the concern in Reference [MHI03]. The pressure in the US-APWR advanced accumulator when nitrogen gas begins to flow out, was lower than the primary system pressure during the SBLOCA, even for the maximum break size case. Therefore, MHI concluded that, for the US-APWR SBLOCA analyses, the extended Henry-Fauske model had not been used. The presence of non-condensable gases in the primary system is limited to the advanced accumulators, and sufficient liquid remains in the advanced accumulators to preclude non-condensable gas from leaving any of the advanced accumulators. Therefore, the staff limits the use of M-RELAP5 for US-APWR SBLOCAs to accidents that do not result in the injection of nitrogen gas into the RCS.

The staff finds the implementation of the Moody critical flow model into M-RELAP5 acceptable and consistent with technical reference [TEC28].

The staff concludes that M-RELAP5 satisfies the Appendix K requirement for I.C.1b, “Discharge model,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.6 Critical Heat Flux and Post-CHF Heat Transfer Model

Appendix K, item I.C.4, requires for CHF correlations that “the computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their representative authors.”

Appendix K, item I.C.5, requires that post-CHF heat transfer correlations “predict values of heat transfer coefficient equal to or less than the mean values of applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used.”

MHI uses the existing RELAP5-3D wall heat transfer correlations in M-RELAP5 for the US-APWR SBLOCA analyses. These models are described in Section 4.2, “Wall-to-Fluid Heat Transfer,” in the RELAP5-3D, “Models and Correlations,” manual, [TEC05]. The correlations for CHF, transition boiling, film boiling and vapor convection heat transfer are all important for analysis.

Critical Heat Flux

The CHF correlation is used to predict CHF occurrence. It is also used to determine when transition boiling heat transfer occurs and the magnitude of the transition boiling heat transfer coefficient.

The 1986 AECL-UO Critical Heat Flux Lookup Table [TEC30] was used in RELAP5-3D and was maintained in M-RELAP5. The CHF table is based on the Chalk River Nuclear Laboratories' CHF data bank, which contains more than 15,000 CHF data points for water. The table method was not designed to replace ad hoc CHF correlations derived for a given geometry over a narrow range of flow conditions; rather it was designed to provide a reasonable estimate of CHF over a very wide range of flow conditions and geometries, especially over the ranges where CHF data are scarce.

The table look-up CHF value is multiplied by a series of eight correction factors to account for differences in the test data characteristics and the actual geometry being modeled. MHI implemented changes to two of these factors. The bundle factor, k2, is modified early in the transient to prevent CHF when it is not expected to occur. A factor, k7', was added to the vertical flow factor, k7, because the RELAP5-3D model did not predict ROSA/LSTF test SB-CL-18 CHF condition.

The bundle factor, k2, is used to account for the effect of the geometry difference between a tube and a rod bundle, the effect of an unheated wall, and the effect of the enthalpy distribution in the bundle due to the rod power distribution. The bundle factor is a function of the equilibrium quality. The equilibrium quality used in the code for wall heat transfer is based on the phasic enthalpies and the mixture enthalpy, with the mixture enthalpy calculated using the flow quality.

The vertical flow multiplier, k7, is treated differently than the other multipliers. In the low mass flux range, k7 is applied. When interpolation is required, the low mass flux range of the interpolation box is multiplied by k7, but the high mass flux range is not.

The RELAP5-3D developers compared the Atomic Energy of Canada University of Ottawa (AECL-UO) lookup table with the tube critical heat flux (CHF) data in the INEL bank [TEC31]. There were 9,687 CHF data points; however some of the data showed energy balance problems and those data were removed for the comparison. The remaining 9,353 CHF data points were compared with the AECL-UO lookup table. The average error, (predicted value minus measured value)/(predicted value), was -0.049 with a root mean square error of 0.39, that is 37 percent.

The staff, in RAI 7-11 [NRC03], requested that MHI address the continued use of the 1986 version since a newer 2006 version [TEC32] is available. MHI addressed this in Reference MHI04. CHF occurrence might be expected in the early stage of the transient before the reactor trips, and during loop seal clearance or boiloff periods when the core uncovers during SBLOCAs. The differences of the CHF predicted by the 1986 version and the 2006 version and its effects on the SBLOCA calculation results were evaluated by MHI for the early stage of the transient and the core uncovery periods, respectively.

During the early stage of the SBLOCA, before reactor trip, the 2006 version results in a 5 percent lower value for the CHF than the 1986 version. MHI attributed this to the difference between the 1986 and the 2006 definition of the hydraulic correction factor, k1.

Representative core pressures of 12 MPa (1740 psia) and 14 MPa (2031 psia) and representative mass velocities of 1500 kg/m^2-s (307 lbm/ft^2-s)and 2500 kg/m^2-s (512 lbm/ft^2-s) in the early stage of the transient, before reactor trip, when CHF occurrence might be expected, were evaluated by MHI. The quality where CHF was expected to occur was less than 0.2 in the early stage of the transient. The CHF values were almost the same when the quality was less than of 0.2. Therefore, the 2006 version could result in a CHF value 5 percent less than the 1986 version because of the difference in the hydraulic correction factor. Since the margin to DNB was greater than 5 percent in the early stage of the transient, DNB would not be expected to occur before reactor trip even if the 2006 version were used, and the calculation results would not be affected. Therefore, MHI concluded continued use of the 1986 version remains acceptable for the early stage of the US-APWR SBLOCA analyses. Based on this evaluation, the staff concludes that the 1986 version is acceptable and conservative for the US-APWR SBLOCA analyses for the early stage of the transient, before reactor trip.

In the US-APWR SBLOCA analysis, CHF occurs during the following: the loop seal clearance period for the 7.5-inch break and during the boiloff period for the 1.0 ft^2 break, at low flow rate conditions in the upward direction, at a pressure of about 9 MPa (1305 psia) and a mass velocity below 20 kg/m^2-s (4 lbm/ft^2-s) for the 7.5-inch break, and at a pressure of about 3 MPa (435 psia) and a mass velocity of about 20 kg/m^2-s (4 lbm/ft^2-s) for the 1.0 ft^2 break. MHI compared the CHF values from the 1986 version and the 2006 version for these conditions.

In the 1986 version, the CHF for low flow conditions is calculated using the CHF value for pool boiling conditions and the vertical flow factor, k7. A comparison of the 1986 look-up table and the Zuber correlation for pool boiling [TEC33] over the pressure range of interest showed the CHF for the two models were in agreement. However, the 2006 look-up table CHF values were considerably smaller than the values from the 1986 version or the values calculated with the Zuber correlation even though the 2006 paper stated that the Zuber correlation is used. The reason for this difference was not described in the paper and the definition of the vertical flow factor, k7, was not included in paper. Therefore, MHI obtained the CHF values at low flow conditions for the 2006 version directly from the look-up table by interpolation rather than using the CHF value at the pool boiling condition and the vertical flow factor, k7. Since the flow regime was considered to be representative of a pool boiling heat transfer mode at low flow conditions, the static quality was used to obtain CHF values from the look-up table instead of the equilibrium or flow quality. MHI compared these values to the 1986 values at the expected pressure and mass velocities when CHF was expected to occur.

During the loop seal clearance or boiloff periods, the 1986 version resulted in CHF values equal to or less than the 2006 version at the quality conditions where CHF was expected to occur. The 1986 version of the AECL-UO look-up table is more conservative during the loop seal clearance and the boiloff periods. Therefore, MHI concluded continued use of the 1986 version remains acceptable for the loop seal clearance or boiloff periods of the SBLOCA analyses. Based on this evaluation the staff concludes that the 1986 version is acceptable and conservative for the US-APWR SBLOCA analyses for the loop seal and boiloff periods.

MHI has included a number of changes to the AECL-UO CHF model:

1. The CHF value in the low mass flux region was reset to the value from the Zuber pool-boiling CHF correlation, and was multiplied by the vertical flow factor to account for the effect of void fraction.

In its response to staff RAI 7-18 [MHI05], MHI clarified the use of the Zuber pool boiling model in Reference MHI05. The amount of CHF data in the AECL-UO database at low flow and two-phase flow condition was limited. Therefore, CHF values from the Zuber correlation were inserted into the CHF table where the mass velocity was 0.0 and the quality was 0.0 for each pressure condition. The CHF value for two-phase conditions is then calculated by multiplying the value taken from the Zuber correlation by the vertical flow factor, k7. This is consistent with the use of the look-up table and is acceptable to the staff.

However when this model was applied to the ROSA-IV/LSTF Test SB-CL-18, CHF was not predicted during the loop seal clearance period. Therefore, an additional multiplier, k7', was introduced by MHI to adjust the CHF at high void fractions.

2. An additional multiplier factor, k7', to the CHF was implemented in M-RELAP5 by MHI to further reduce the CHF at high void fractions and low mass flux conditions based on comparison of the CHF correlations to the ROSA-IV/LSTF experimental data (See Section 8.2.1, “ORNL/THTF Void Profile and Uncovered-Bundle Heat Transfer Tests,” in Topical Report MUAP-07013-P).

MHI further addressed the additional multiplier in its response to RAI 8.2.1-20 [MHI07]. The factor k7’ is used to improve the ability to predict CHF occurrence during the loop seal clearance phase and during the boiloff phase when the flow and void fraction conditions are met. The multiplier is required to account for the non-uniform flow distribution effect in the bundle during the loop seal clearance phase, where liquid flows down into the core from the upper plenum. As during the boiloff phase, the region above the core is nearly filled with vapor and there is little liquid available to flow down into the core from the upper plenum, the multiplier is not necessary during the boiloff phase. However, the use of the multiplier gives more conservative results and is used for all phases when the flow and void fraction conditions are met. The staff finds the use of the k7’ multiplier acceptable for use throughout the transient because it provides additional conservatism in the CHF prediction.

3. When the AECL-UO look-up table is used to calculate the CHF, M-RELAP5 has a special logic in use of the bundle factor multiplier, k2, when certain conditions are met. With the bundle factor, MHI believes there is an unrealistically large decrease in the CHF value. Control logic to apply the bundle factor, k2, in the CHF calculations has been implemented in M-RELAP5.

Applicability of the look-up table to bundle geometry

MHI did not clearly describe the technical basis for this logic. The staff, in RAI 7-10 [NRC03], requested that MHI provide the technical analysis and benchmark basis. MHI provided its response to this RAI in Reference MHI04. DNB would occur early in the transient, when the bundle factor is applied for both the 7.5-inch break and the 1.0- ft^2 break, because the bundle factor greatly reduces the CHF. MHI noted that the bundle factor is based on a limited amount of data that is not specified in the AECL-UO paper. MHI compared the AECL-UO look-up table CHF value, with the modified bundle factor, to its WRB-2 thermal design correlation during the early period of the transient, before reactor trip. The WRB-2 correlation predicts a larger CHF value, both for a typical cell and a thimble cell (which has a slightly larger CHF than a typical cell). For both breaks, the calculated heat flux early in the transient was well below both CHF values. MHI concluded that DNB does not occur early in the transient and the analyses with the modified bundle factor early in the transient are conservative. Based on this evaluation, the staff finds the use of the modified bundle factor in M-RELAP5 is acceptable.

The staff, in RAI 7-22 [NRC04], requested that MHI address the range of break sizes for the US-APWR, which requires the use of the modified bundle factor. MHI provided its response in Reference MHI06.

MHI extended the previous study for the 7.5-inch and 1.0- ft^2 breaks to the full small-break spectrum by including six additional break sizes. The same bounding analyses were performed for a break in the cold leg for the other break sizes. MHI compared the CHF calculated by the WRB-2 correlation, which was developed for the US-APWR fuel assembly, to the 1986 CHF look-up table with the modified bundle factor. The WRB-2 correlation predicted a larger CHF value than the AECL-UO look-up table with the modified bundle factor for all break sizes, confirming the AECL-UO look-up table with the modified bundle factor can be conservatively used to determine if DNB will or will not occur early in the transient for all break sizes.

MHI also assessed other break locations. The DNB margin decreases as the break size increases. The break flow reduces the RV inlet flow in the broken loop, which then reduces the core flow. The reduced core flow reduces the DNB margin. Since the break flow is largest for 1.0-ft^2 break, the DNB margin is the least for 1.0-ft^2 break. A break in the crossover leg, pump suction side, would be less limiting since the pressure in this region is lower than in the cold leg, pump discharge side, due to the pump head. The lower pressure in a crossover leg leads to a lower break flow than that through a cold-leg break with the same break size. Therefore, the DNB margin for a crossover leg break is greater than that for a cold-leg break. Since the break flow does not adversely affect the core flow for a hot-leg break or a pressurizer steam-phase break, the DNB margins for these break locations are greater than that for the cold-leg break. The break size for a DVI-line break is less than 1.0-ft^2; therefore, the DNB margin for this break is greater than for the1.0-ft^2 cold-leg break.

Since the CHF value predicted with the AECL-UO look-up table with the modified bundle factor is greater than the expected rod surface heat flux for all break sizes, MHI concluded that DNB does not occur early in the transient for all break sizes. The modified bundle factor is therefore used in all US-APWR SBLOCA analyses. Based on these evaluations the staff finds the use of the modified bundle factor acceptable for all SBLOCA breaks.

In Request 7-22 [NRC04], MHI was also asked to explain why the AECL-UO look-up table with the modified bundle factor is applicable to the 14-ft US-APWR fuel assembly. MHI provided its response in Reference MHI06.

The maximum heated length of the experimental data used to development the AECL-UO look-up table was19.7 ft; therefore, MHI concluded the AECL-UO look-up table was applicable to the 14-ft US-APWR fuel assembly. The WRB-2 correlation for the US-APWR fuel assembly has been validated with DNB experiments using a heated length test section of 12 ft. MHI believes that the WRB-2 correlation is applicable to the US-APWR fuel assembly, since the dominant parameters for DNB in a PWR fuel assembly are the grid structure and grid axial spacing, not the heated length. MHI intends to perform additional DNB experiments using the test sections of 14-ft heated length for further confirmation of the applicability of the WRB-2 correlation to the US-APWR fuel assembly to accommodate the staff's suggestions [MHI36].

Applicability of the look-up table to a flow channel with an unheated wall

The CHF value was also calculated with the WRB-2 correlation for the flow channel with an unheated control rod guide thimble wall using the geometry parameters of the thimble cell for

the US-APWR fuel assembly. The mass velocity was decreased ten percent to account for the difference of the hydraulic diameter between the thimble cell and fuel assembly. The CHF value calculated by WRB-2 was larger than that obtained from the look-up table and also slightly larger than that for the typical cell. MHI concluded the look-up table conservatively predicts CHF of the thimble cell with the modified bundle factor. Based on this evaluation the staff finds the use of the modified bundle factor acceptable for a flow channel with an unheated wall.

Evaluation of the effect of the enthalpy distribution in the fuel assembly on the CHF

The rod power distribution in the fuel assembly results in an enthalpy distribution in the fuel assembly. The subchannel with the highest enthalpy usually reaches DNB before a subchannel with average conditions. MHI believes that the main purpose of the bundle factor was to account for the effect of the enthalpy distribution in the fuel assembly on CHF. Conservative calculations were performed, by MHI, to justify using the modified bundle factor. The hot rod power was applied to the entire hot assembly in the M-RELAP5 calculations to conservatively account for the effect of the enthalpy distribution in the fuel assembly. The CHF value obtained from the look-up table with the modified bundle factor was greater than the actual heat flux. Based on this evaluation, the staff finds the use of the modified bundle factor acceptable for the evaluation of the effect of the enthalpy distribution in the fuel assembly.

The applicability of the AECL-UO lookup table, as modified by MHI to CHF predictions, was validated by comparisons to test data, as described in Section 8.1.2, “ORNL/THTF Void Profile and Uncovered-Bundle Heat Transfer Tests,” in Topical Report MUAP-07013-P.

In April 2010, a member of the International RELAP5-3D User’s Group reported an error in the RELAP5-3D implementation of the AECL-UO CHF lookup table after comparing the coding to the 1986 paper. A subsequent, more detailed review by RELAP5-3D developers revealed 41 errors or inconsistencies, some of which were minor (e.g., difference in the last digit due to round off for interpolated values), and some of which were significant. These errors affected all RELAP5 versions including RELAP5/MOD3 version 3.0 and all previous versions of RELAP5-3D and M-RELAP5. MHI described the RELAP5 implementation of the AECL-UO lookup table and provided the context for understanding the significance of the errors in its response to Request 2 in Reference [MHI37].

MHI modified the M-RELAP5 code and produced a new version, Version1.6, which corrected all the errors. An assessment of the impact of these changes showed the PCT for the limiting US-APWR SBLOCA (1.0 ft^2 cold-leg break) decreased by about 11 ºF (this assessment also included the mass conservation correction described in Section 4.6.7, “Advanced Accumulator Model,” of the SE). The DCD analyses for the US-APWR SBLOCA evaluation include the error corrections.

The staff concludes that M-RELAP5 satisfies the Appendix K requirement for I.C.4, “Critical heat flux,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs. Employing the modified bundle factor prevents the unrealistic prediction of CHF, which would occur if the original bundle factor was applied, is acceptable to the staff. The CHF model in M-RELAP5 used for the US-APWR is conservative with respect to the fuel design WRB-2 CHF correlation.

Transition Boiling Heat Transfer

The transition boiling heat transfer model incorporated in RELAP5-3D and used in M-RELAP5 is based on the Chen transition boiling model [TEC34]. This model considers the total transition

boiling heat transfer to be the sum of individual components, one describing wall heat transfer to the liquid (boiling term) and a second describing the wall heat transfer to the vapor (convective term). The Chen transition boiling model was compared to 4,167 data points from eight sources for water flowing in tubes. The mean deviation of the measured heat flux to the predicted heat flux is 16.0 percent.

Film Boiling Heat Transfer

Film boiling heat transfer consists of conduction across the vapor film blanket next to a heated wall, convection to the flowing vapor and radiation across the film to a continuous liquid blanket or a dispersed mixture of liquid droplets and vapor. The conduction heat transfer coefficient through the vapor film incorporated in RELAP5-3D, and used in M-RELAP5, is the Bromley correlation [TEC11]. The model was shown to correlate within ± 18 percent of the available data. The conductive portion of the total experimental heat flux was obtained by calculating and subtracting a radiation component based on a parallel plate model using an appropriate wall and liquid emissivity.

The transition boiling and film boiling heat transfer models are used to predict the cladding temperature behavior during the core mixture level recovery period. Applicability of these models for the heat transfer during the core mixture level recovery period was validated with test data, as described in Section 8.1.3, "ORNI/THTF High-Pressure Reflood test," in Topical Report MUAP-07013-P. MHI concluded that the transition boiling and film boiling heat transfer models incorporated in M-RELAP5 predicted the experimental data in an acceptable manner. The staff agrees and finds the transition and film boiling heat transfer correlations acceptable for use in M-RELAP5 for the US-APWR SBLOCA analyses.

The staff concluded that M-RELAP5 satisfies the Appendix K requirement for I.C.5, "Post-CHF heat transfer," for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

Vapor Convection Heat Transfer

The rod wall heat transfer above the two-phase mixture level is important to the evaluation of the PCT in SBLOCA analysis. The heat transfer just above the two-phase mixture level depends on film boiling at high quality condition where the vapor convection term is dominant. The heat transfer depends on single vapor convection above the two-phase flow region. For heat transfer from a heated wall to a single-phase vapor for turbulent forced convection, MHI uses the Dittus-Boelter correlation [TEC21]. In two-phase flow, the liquid mass flux times the vapor-to-liquid density ratio is added to the vapor mass flux. This effectively converts the Dittus-Boelter correlation to a two-phase vapor convection heat transfer model and a smooth transition from two-phase flow to single-phase vapor flow is modeled in M-RELAP5.

This converted correlation for the two-phase vapor convection heat transfer model is implemented in RELAP5-3D, and M-RELAP5, with the Dougall-Rohsenow correlation [TEC35] except that the physical properties of the vapor, the thermal conductivity, viscosity and specific heat, are evaluated at the film temperature. The saturation temperature was used to evaluate the physical properties of the vapor in the original Dougall-Rohsenow correlation.
The use of the modified correlation results in a smaller heat transfer coefficient when compared to the original correlation.

Appendix K states that: "…use of the Dougall-Rohsenow correlation under conditions where non-conservative predictions of heat transfer result will no longer be acceptable." ORNL rod

bundle data show that the original Dougall-Rohsenow correlation overpredicts heat transfer for high quality conditions. This result was obtained using the fluid saturation temperature rather than the actual vapor temperature for the vapor temperature. M-RELAP5 is capable of calculating the actual vapor temperature with its non-equilibrium model. Therefore, MHI expects M-RELAP5 can satisfactorily calculate the rod heat transfer from two-phase vapor convection with the modified Dougall-Rohsenow correlation.

The applicability of the Dittus-Boelter correlation and the modified Dougall-Rohsenow correlation for SBLOCA analyses was validated with experimental data, which simulated the core uncover phase during a SBLOCA. The results of the comparison with high pressure test data were presented in Section 8.1.2, “ORNL/THTF Void Profile and Uncovered-Bundle Heat Transfer Tests,” of Topical Report MUAP-07013-P. The staff evaluation and acceptance of these tests is discussed in Section 4.4.1.2, “ORNL/THTF Void Profile and Uncovered-Bundle Heat Transfer Tests,” of this SE. The results of the comparison with low pressure test data were presented in Section 8.1.4, “FLECHT-SEASET Forced-Reflood Test,” of Topical Report MUAP-07013-P. The staff evaluation and acceptance of these tests in discussed in Section 4.4.1.4, “FLECHT-SEASET Forced-Reflood Test,” of this SE.

MHI concluded the vapor convection heat transfer model, as implemented in M-RELAP5, predicted the experimental data reasonably well. The staff agrees and finds the vapor convection heat transfer model acceptable and conservative for use in M-RELAP5 for the US-APWR SBLOCA.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.C.5, “Post-CHF heat transfer,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

Prevent Return to Nucleate Boiling and Transition Boiling

Appendix K, item I.C.4e, requires that the return to nucleate boiling be prevented during the blowdown phase once CHF has been predicted. The original RELAP5-3D did not contain any logic to prevent return to nucleate boiling once CHF had occurred. A new heat transfer control parameter, to prevent the return to nucleate boiling during the blowdown after CHF first occurrence, has been added in M-RELAP5 by MHI to satisfy this Appendix K requirement.

Appendix K, item I.C5b, requires that the returns to transition boiling be prevented during the blowdown after the cladding surface superheat exceeds 300 ºR. The original RELAP5-3D did not contain any logic to prevent return to transition boiling. The new heat transfer control parameter noted above is also used to prevent the return to transition boiling once the cladding superheat has exceeded 300 ºR.

MHI implemented a prevention logic scheme in M-RELAP5 to comply with these requirements.

Applicability of the transition boiling and nucleate boiling models, incorporated in M-RELAP5 for the heat transfer and rewet phenomena during the core mixture level recovery phase, was validated with a comparison to test data as shown in Section 8.1.3, “ORNI/THTF High-Pressure Reflood test,” in Topical Report MUAP-07013-P.
MHI interpretations of the Appendix K requirements concluded these requirements were only necessary during the blowdown phase of the accident and were not necessary during the core mixture level recovery period.

The staff, in RAI 7-12 [NRC03], requested that MHI address its interpretation of these Appendix K requirements and to clarify the definition of the blowdown phase. MHI provided its response in Reference MHI03. MHI divides the SBLOCA transient into five periods: blowdown, natural circulation, loop seal clearance, boiloff and core recovery. The blowdown period starts from the initiation of the break. It ends when the primary system pressure has decreased to nearly equal to the secondary system pressure. The blowdown period is generally followed by the natural circulation and loop seal clearance periods during SBLOCA transients.

It was difficult to define the end of blowdown period for larger break sizes (larger than a 10-inch break for the US-APWR), as the primary pressure rapidly decreases below the secondary pressure and the natural circulation period and loop seal clearance periods were nonexistent. For these breaks, the prevention logic is applied until the core recovery period, when the primary inventory begins to increase from the accumulator injection.
Even if CHF or excess cladding overheating occurs in the early stage of the blowdown period, the return to nucleate boiling or the return to transition boiling can be expected soon after the reactor trip because the rod surface heat flux is greatly reduced and sufficient coolant exists in the core. However, these rewetting phenomena were not validated by any experimental data. Therefore, the prevention logic of the Appendix K requirement is conservatively applied during the blowdown period in the SBLOCA analysis. The staff finds the use of the prevention of return to transition boiling during the blowdown period acceptable.

Return to nucleate boiling after CHF or return to transition boiling after excess cladding overheating during loop seal clearance, or core recovery periods due to the core mixture level recovery, was validated using the ORNL/THTF high pressure reflood tests and the ROSA-IV/LSTF test. Therefore, the prevention logic is not necessary for these periods. Based on these assessments, the staff agrees that it is not necessary to prevent return to transition boiling during these periods.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.C.4e, “Return to nucleate boiling,” and item I.C.5b, “Return to transition boiling,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.7 Advanced Accumulator Model

Appendix K, item I.D.3, requires the reflood rate to be calculated by an acceptable model that takes into consideration the thermal hydraulic characteristics of the core and the reactor coolant system.

MHI added a model for the advanced accumulator into M-RELAP5. The total resistance coefficient, or pressure loss from the accumulator exit to the RCS, is determined from the accumulator flow rate coefficient and the resistance coefficient from the injection piping.

The accumulator flow rate coefficient is a function of a cavitation factor and the water level in the accumulator. The accumulator flow rate coefficient is calculated from empirical correlations obtained from test data which covered the range of applicability for the US-APWR design. The accumulator flow rate empirical coefficients were derived separately for both the large and small flow rate injections as a function of the cavitation factor. The advanced accumulator model is described in Appendix D in Topical Report MUAP-07013-P.

In response to Request D-2 [MHI05], MHI clarified the advanced accumulator model, as incorporated into M-RELAP5. The US-APWR advanced accumulator is modeled as cylindrical

in the vertical direction. For the cylindrical type accumulator, Topical Report MUAP-07013-P equations D-16 to D-19 and D-34 to D-35 are not used. The liquid head in the accumulator tank is calculated by Topical Report MUAP-07013-P equation D-23; therefore it is not necessary to explicitly express the liquid level in the tank in calculating the cavitation factor.

The engineered safety features in the US-APWR design allow the presence of non-condensable gas in the accumulator tank for the flow calculation, but prevents it from entering the RCS during SBLOCAs. Therefore, the capability of M-RELAP5 to predict the effect of non-condensable gas is not directly addressed in the Topical Report MUAP-07013-P.

In response to Request E-1 [MHI07], MHI acknowledged that M-RELAP5 will be further reviewed by the staff in the future if the code is applied for analyses that need to account for the effect of non-condensable gas.

The advanced accumulator injection characteristic was validated against full height 1/2-scale injection experiments. M-RELAP5 analysis results of the injection volumetric flow rate, the tank pressure and the tank water level for four cases were compared to the test results. In each case, MHI concluded the analysis results were in good agreement with the test results, and the injection characteristic was well simulated by the advanced accumulator model. In particular, the analysis results satisfactorily reproduced the test results for the tank water level, which is the integration value of the injection volumetric flow rate.

The total uncertainty of the experimental equation for the accumulator flow rate coefficient used for the safety analysis of the US-APWR was derived from instrument uncertainty, data scatter and manufacturing tolerances. The staff, in Request D-2 [NRC04], requested that MHI clarify the uncertainties used for the accumulator flow rate coefficient. As indicated in Reference MHI03, the uncertainty analysis is under review by the staff as part of its review of the advanced accumulator design report [MHI38].

The completion of the accumulator uncertainty analysis review does not impact the review of the advanced accumulator model; however, these uncertainties are considered in the US-APWR safety analysis to determine the advanced accumulator flow rate.

MHI modified the M-RELAP5 code and produced a new version, Version1.6, to improve the numerical scheme for conserving the mass in the accumulator component. MHI found that the use of a liquid volume balance instead of a mass balance caused a small mass error in the accumulator component, as described in its response to Request 3 in Reference [MHI37].

An assessment of the impact of this change showed the PCT for the limiting US-APWR SBLOCA (1.0 ft^2 cold-leg break) decreased by about 11 ºF (6 °C) (this assessment also included the AECL CHF correction described in Section 4.3.3.6, “Critical Heat Flux and Post-CHF Heat Transfer Model” of the TER).

The staff finds the model implemented in M-RELAP5 for the advanced accumulator acceptable for use in the evaluation of the US-APWR ECCS performance for SBLOCAs.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.D.3, “Reflood rate” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs. A model for the advanced accumulator, based on scaled testing, has been incorporated into M-RELAP5, by MHI, for the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.8 Appendix K Requirement for ECCS Bypass

MHI believed the Appendix K item I.C.1c requirement for ECC water bypass was not applicable to the US-APWR SBLOCA.

MHI noted the staff has previously addressed the end of bypass in the staff's SER for the Westinghouse NOTRUMP code [TEC36], Table VIII-1, "NOTRUMP Conformance with Appendix K to 10 CFR 50," dated April, 18, 1985). The staff concluded that this requirement was not applicable to SBLOCAs. However, during this review the staff noted that the break size range used by MHI for the US-APWR SBLOCA analyses was larger than the range considered in the NOTRUMP review and it was not clear if the previous conclusion was applicable to the 1.0-ft^2 break. The staff, in Request 7-14 [NRC04], requested that MHI provide results from the M-RELAP5 calculations for the 7.5-in and 1.0-ft^2 breaks for a number of parameters, to assist the staff in resolving issues related to end of bypass. MHI provided the requested results, in graphical form, in Reference MHI05.

The staff reviewed the results for the 1.0-ft^2 break, presented in DCD Tier 2, Section 15, Figures 15.6.5-23 through 15.6.5-31, and the results provided in response to Request 7-14, Figures RAI-7-14.14 through RAI-7-14.16 [MHI05]. For the 1.0-ft^2 break, accumulator injection starts at 90 sec and SI injection starts at 126 sec. The upper core region uncovery starts at 103 sec and recovers at 326 sec. The event timing is provided in DCD Tier 2 Table 15.6.5-11, "Sequence of Events for 1-ft^2 Small Break LOCA."

As seen in Figure RAI-7-14.14, "Cold-leg to Downcomer Mass Flowrates for 1-ft^2 Break," the ECC injected by the accumulators flowed around the downcomer and out the break. In addition, as seen in DCD Tier 2 Figure 15.6.5-27, "RCS Mass Inventory for 1-ft^2 Small Break LOCA," the RCS inventory continued to decrease until the SI pumps began to deliver ECC. DCD Tier 2, Figure 15.6.5-28, "Downcomer Collapsed Level for 1-ft^2 Small Break LOCA," showed the downcomer did not empty, so there was always some liquid that prevented/delayed large amounts of steam from exiting the bottom of the core and holding up the ECC injected liquid. This was also shown in Figure RAI-7-14.17, "Average Core Entrance and Exit Vapor Mass Flowrates for 1-ft^2 Break," where there was essentially no steam flow out of the bottom of the core into the downcomer that could cause bypass of the injected ECC flow. The vapor mass flow rate was into the bottom of the core rather than out of the bottom of the core. The same was true of the liquid flow and, thus, no potential for flow up the downcomer. Figure RAI-7-14.15, "Downcomer to Lower Plenum Mass Flowrate for 1-ft^2 Break," showed that the downcomer to lower plenum mass flow became positive, i.e. downward flow in the downcomer, before the accumulators started to inject.
The bypass flow between the upper head and the downcomer could potentially provide a path for steam to enter the downcomer during the loop seal clearance period. When the coolant loops are sealed at the crossover legs during the loop seal clearance period, the upper head/downcomer bypass flow path has the potential to relieve vapor generated in the core. Only the high head SI system injects ECC water during the loop seal clearance period.

In staff RAI 7-13 [NRC04], MHI was asked if the bypass flow path was modeled and requested that MHI address the effect of the bypass flow path between the upper head and the downcomer. As discussed in Reference MHI05, the bypass flow path between the upper head and the downcomer is considered in the US-APWR SBLOCA analyses. However, MHI's assessment concluded the effect of the bypass on the PCT or on the sweep out of ECC water out the break was small, based on the following evaluation.

For the 7.5-inch break, the loops are sealed at the crossover legs during the loop seal clearance period, and the bypass flow path between the upper head and the downcomer has the potential for relieving vapor generated in the core. If the vapor is relieved through this bypass flow path, the pressure above the core is reduced and the core liquid level depression is reduced. The reduction of the liquid level depression lowers the cladding temperature when the core is uncovered so that there is a possibility that the bypass flow path affects the PCT during the loop seal clearance period. However, a large amount of liquid remains in the upper head, and the bypass flow path is covered by the liquid during the loop seal clearance period for the break sizes in which the core is uncovered. Since there is essentially no vapor relief through the bypass flow path during the loop seal clearance period, the PCT is not likely to be affected by the bypass flow path.

When the liquid level in the upper head is lower than the top of the spray nozzles, the vapor generated in the core flows directly into the upper downcomer and from there it flows toward the broken cold leg. Since the ECC water from the high head SI systems is injected into the downcomer in the downward direction, the vapor flow from the spray nozzles does not impede this ECC water flow into the core. For the larger break sizes, the accumulator injects ECC water into the cold legs when the primary pressure decreases. The vapor from the intact steam generator sweeps part of this ECC water out the break. There is a chance that the vapor from the spray nozzles also sweeps part of the ECC water out the break. For larger breaks, MHI considered the 1.0 ft^2 break. The vapor flow from the spray nozzles was negligible compared with the vapor flow from the intact loop, and its effect on the ECC water is expected to be small.

Sensitivity analyses were performed by MHI for the US-APWR [MHI02]. The results of these studies were provided to further investigate ECC bypass.

The effect of the break orientation (top, side or bottom) on the RCS behavior were small, and the effects on the loop seal clearance period, the ECC water injection start time, and the collapsed liquid level in the vessel upper head were small. Loss of ECC water from steam flow did not occur in the sensitivity calculations. Loop seal formation does not occur for hot leg, crossover leg and pressurizer steam phase breaks, and a loss of ECC water from steam flow will not occur. ECC water from only the high head SI system is injected during the loop seal clearance period for the DVI injection line break, and a loss of ECC water from steam flow did not occur.

None of the sensitivity analyses for the noding and time step size studies showed loss of ECC water from steam flow because the sensitivities to these parameters were small.

Sensitivity analyses with a no single failure assumption and with offsite power available were also performed. ECC water injection did not occur during the loop seal clearance period for the no single failure assumption, and only ECC water from the high head SI systems was injected during the loop seal clearance period for the case with offsite power available. Loss of ECC water from steam flow did not occur in both cases.

Based on the information provided, the staff finds ECC entrainment due to steam upflow in the downcomer will not occur during a SBLOCA in the US-APWR. No special considerations, like tracking the amount of ECC water entering the RV with downcomer upflow and artificially removing this inventory from the vessel once upflow ceases, are needed.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.C.1c, “ECCS water bypass,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs. Based

on the SBLOCA response of the US-APWR, as designed and described in the DCD, no special model is needed to account for this requirement for the evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.9 Appendix K Requirement for Refill/Reflood Heat Transfer

MHI believed the Appendix K requirement item I.D.6 for refill/reflood heat transfer was not applicable to the US-APWR SBLOCA. However, there could be core uncovery resulting in cladding superheat. The staff, in RAI 7-2 [NRC03], requested that MHI confirm that all SBLOCA cases do not require refill/reflood heat transfer, however if refill/reflood heat transfer is needed then MHI should explain why this Appendix K requirement is not applicable to the US-APWR SBLOCA.

MHI responded to this request in Reference MHI03. Appendix K requires that an applicable FLECHT heat transfer correlation be used for reflood rates of one inch per second or higher. It also requires, when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only to steam, and shall take into account any flow blockage due to cladding swelling or rupture. MHI further stated since "As a FLECHT heat transfer correlation is not used in M-RELAP5, this requirement is taken as not applicable to SBLOCA. But, additional form loss coefficients due to flow blockage are applied in M-RELAP5 when a fuel rod ruptures in accordance with the requirement." MHI also stated "The reflood heat transfer after the core uncovers is calculated by the heat transfer package incorporated in M-RELAP5, and it is validated by ORNL/THTF high-pressure reflood tests."

Flow blockage due to cladding swelling or rupture is accounted for in M-RELAP5 as discussed in Section 4.6.4 of this SE.

MHI noted the staff has previously addressed use of the FLECHT heat transfer correlation for reflood rates greater than one inch per second in the staff's SER for the Westinghouse NOTRUMP code [TEC36]. The staff concluded that this requirement was not applicable to SBLOCAs. An appropriate heat transfer correlation is acceptable to the staff. The acceptability of the M-RELAP5 heat transfer model is discussed in Section 4.6.6, *Vapor Convection Heat Transfer*, of this SE.

Reflood Rates less than 1 in/sec

During this review the staff noted that the break size range used by MHI for the US-APWR SBLOCA analyses is larger than the range considered in the NOTRUMP review.

The staff, in RAI 7-14 [NRC04], requested that MHI provide results from the M-RELAP5 calculations for the 7.5-in and 1.0-ft^2 breaks, for a number of parameters, to assist the staff in resolving issues related to the FLECHT heat transfer applicability. MHI provided this information in Reference MHI05.

For the 7.5-in break, the hot assembly two-phase water level decreased about 5.5 ft into the core at 122 sec as shown in Figure RAI-7-14.10 [MHI05], "Core Two-Phase Level for 7.5-inch Break." The SI started to flow at 130 sec and the level recovered at 143 sec. The event timing was provided in DCD Tier 2 Table 15.6.5-9, "Sequence of Events for 7.5-inch Small Break LOCA." This indicated a reflood rate of about 5 in/sec for the 7.5-inch break. MHI also provided the estimated reflood rate for the core recovery phase of the limiting 1.0-ft^2 break. The estimated reflood rate for this case was greater than 1 in/sec, ranging from about 3 to 5 in/sec.

Therefore, MHI concluded its position on the Appendix K requirement for steam cooling for reflood rates less than 1 in/sec, which does not apply to the SBLOCA analyses, was confirmed for the larger break sizes.

A model to address the Appendix K requirement related to cooling to steam only for reflood rates less than 1 in/sec is not required for the US-APWR SBLOCA analysis with M-RELAP5.

The staff concludes M-RELAP5 satisfies the Appendix K requirement for I.D.6, “Refill/reflood heat transfer,” for use in the evaluation of the US-APWR ECCS performance for SBLOCAs. Based on the SBLOCA response of the US-APWR, described in the DCD, the wall-to-fluid heat transfer models in M-RELAP5 conservatively evaluate the heat transfer during the core heatup and core recovery phases and are acceptable for use in evaluation of the US-APWR ECCS performance for SBLOCAs.

4.6.10 Additional Appendix K Models Considerations

Large, rapid flow oscillations in the vessel, including the core, were seen in some of the figures for the 1.0-ft^2 break provided by MHI in response to RAI 7-14 [MHI05]. This raised the question regarding whether these flow oscillations promote core cooling. Appendix K, Section I.C.7.a, “Core Flow Distribution During Blowdown,” requires that “The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).”

In response to RAI 7-23 [MHI10], MHI addressed the large, rapid flow oscillations in the vessel, including the core. In the US-APWR SBLOCA 1-ft^2 cold leg break, significant oscillations at the core inlet appear after the end of the blowdown phase. The period of the flow oscillations is about 0.2 seconds during the boiloff phase, and is greater than 1 second during the core recovery phase. Therefore, there are no rapid oscillations required to be smoothed as required by Appendix K Section I.C.7.a.

MHI also performed sensitivity calculations to assess the impact of the flow oscillations on the heat transfer in the core. The model represented only the core, the lower plenum, and the upper plenum. Boundary conditions were used for the lower plenum pressure, temperature and void fraction and the flowrate into the core. Boundary conditions were also used for the upper plenum pressure, temperature, and void fraction.

The boundary condition for the core inlet flowrate was taken from the US-APWR DCD 1.0-ft^2 break calculation at 0.5-second intervals which reproduced the significant oscillations observed during the boiloff and core recovery phases. An additional sensitivity case was conducted with a boundary condition that was obtained with a moving-average with a 5 second span on the flowrate for the time-period from 100 seconds after the break initiation. The effect of smoothing the core inlet flow oscillations was negligible and MHI concluded the flow oscillations did not promote core cooling, as shown in Figure RAI-7-23.7, “Peak Cladding Temperature.”

Based on the sensitivity studies performed by MHI, the staff finds there is no need to apply flow smoothing to US-APWR SBLOCA analyses to address Appendix K Section I.C.7.a, “Core Flow Distribution During Blowdown.”

5.0 Conclusions and Limitations

5.1 PIRT and Validation Plan

Based on the review performed, as discussed in Section 4.1 of this SE, the staff finds the development of the US-APWR PIRT followed acceptable practices. The US-APWR is similar to current operating large PWRs and no new processes or phenomena unique to the US-APWR were identified by MHI during its development. The staff agrees that there are no new processes or phenomena unique to the US-APWR.

Based on the review performed, as discussed in Section 4.1 of this SE, the staff finds the PIRT rankings acceptable. The staff finds the validation plan developed by MHI to incorporate appropriate Appendix K models into the M-RELAP5 code acceptable. The staff finds acceptable the validation plan to assess M-RELAP5 against appropriate SETs and IETs for the high ranked processes and phenomena.

Table 5 below provides a summary of the PIRT, including the identification of the high ranked processes and phenomena that are addressed through the implementation of appropriate and acceptable Appendix K models in M-RELAP5, and the SETs and IETs used for the validation of M-RELAP5 for US-APWR SBLOCA analyses.

5.2 Major Modeling Constructs

The nodalization of the US-APWR was developed following the general user guidelines developed for RELAP5-3D. Studies for the break location and orientation were performed in Technical Report MUAP-07025-P [MHI24] and the cold leg, bottom orientation was determined to be the limiting break location for PCT. Nodalization studies near the break and the DVI injection location were performed. Nodalization studies for the SG U-tubes and crossover leg were also performed.

Based on the review performed, as discussed in Section 4.2 of this SE, the staff finds the M-RELAP5 US-APWR nodalization acceptable for the evaluation of SBLOCAs to demonstrate compliance with 10 CFR 50, Appendix K requirements. I.C.1a, I.C.1d, II.2 and II.3.

5.3 Code Modifications

The code modifications made by MHI in the development of the M-RELAP5 code, from the RELAP5-3D code, necessary to comply with 10 CFR 50 Appendix K requirements, are summarized in Section 4.3 of this SE. The staff evaluation of the acceptability of these models is provided in Section 4.6 of this SE.

5.4 Validation

Based on the review performed, as discussed in Section 4.4 of this SE, the staff finds the SETs and IETs used for the validation of M-RELAP5 for US-APWR SBLOCA analyses acceptable for demonstrating the code capabilities for the high-ranked processes and phenomena identified in the US-APWR that are not addressed with Appendix K models.

Table 5. US-APWR SBLOCA PIRT Important Processes and Phenomena Implementation/Validation Summary

Location	Process/Phenomena	Appendix K	Input/Sensitivity	Separate Effects Tests: ROSA/LSTF Void Profile	ORNL/THTF Void Profile	ORNL/THTF Uncovered	ORNL/THTF High-	FLECHT-SEASET Forced-	UPTF Full-Scale SG	Dukler Air-Water	Integral Effects Tests: ROSA-IV/LSTF Small Break	ROSA-IV/LSTF Small Break	ROSA/LSTF Small Break (17%) LOCA	LOFT Small Break (2.5%)	Semiscale Small Break (5%) LOCA
Fuel rod	3. Decay Heat	X													
	7. Local Power		X												
Core	9.CHF/Dryout					X					X	X	X	X	X
	10. Uncovered Core Heat Transfer					X	X	X			X	X	X	X	X
	11. Rewet (Heat Transfer Recovery)						X				X	X	X	X	X
	14. Mixture Level			X	X						X	X	X	X	X
	16. 3-D Power Distribution		X												
Steam generator	37. Water Hold-Up in SG Inlet Plenum								X		X	X	X	X	X
	38. Water Hold-Up in U-Tube Uphill Side									X	X	X	X	X	X
	39. Primary Side Heat Transfer										X	X	X	X	X
	40. Secondary Side Heat Transfer (Level)										X	X	X	X	X
Crossover leg	44. Water Level in SG Outlet Piping										X	X	X	X	X
	45. Loop Seal Formation and Clearing										X	X	X	X	X
Downcomer/Lower Plenum	60. Mixture Level/Void Distribution										X	X	X	X	X
	64. DVI/SI Water/Flowrate		X												

Location	Process/Phenomena	Appendix K	Input/Sensitivity	Separate Effects Tests							Integral Effects Tests				
				ROSA/LSTF Void Profile	ORNL/THTF Void Profile	ORNL/THTF Uncovered	ORNL/THTF High-	FLECHT-SEASET Forced-	UPTF Full-Scale SG	Dukler Air-Water	ROSA-IV/LSTF Small Break	ROSA-IV/LSTF Small Break	ROSA/LSTF Small Break (17%) LOCA	LOFT Small Break (2.5%)	Semiscale Small Break (5%) LOCA
Break	67. Critical Flow	X													
	68. Break Flow Enthalpy		X												

5.5 Scaling

Based on the review performed, as discussed in Section 4.5 of the SE, the staff finds the ROSA-IV/LSTF appropriate and acceptable for use in the validation of the M-RELAP5 capability to model the high-ranked PIRT processes and phenomena that are not addressed with Appendix K models.

5.6 Appendix K Requirements

Based on the review performed, as discussed in Section 4.6 of this SE, the staff finds that M-RELAP5 contains appropriate and acceptable models, as summarized in Section 4.3 of this SE, that comply with the requirements in 10 CFR 50 Appendix K.

5.7 Limitations

1. The M-RELAP5, Version 1.6 code is approved for SBLOCA analysis of the US-APWR only. This approval is based on use of the 1D models in M-RELAP5. Use of the 3-D capabilities of the M-RELAP5 code was not reviewed. Therefore, use of the 3-D capabilities of M-RELAP5 will require further staff review.

2. Presence of non-condensable gases in the primary system is limited to the accumulators, and sufficient liquid must remain in the accumulators to preclude non-condensable gas from leaving any of the accumulators.

3. Modeling of the reactivity feedback effects of boron or other soluble materials in the system is not within the approved capabilities of the M-RELAP5 model.

4. The use of M-RELAP5 for US-APWR SBLOCA with reflood rates less than 1 in/sec will require additional consideration regarding the Appendix K requirement on use of FLECHT limited heat transfer coefficients. Only one assessment was performed for a reflood rate less than 1 in/sec; however, the expected minimum reflood rate for the US-APWR SBLOCA is about 3 in/sec.

5. The use of the AECL-UO CHF lookup table, with a modified bundle factor, is acceptable provided the fuel design specific CHF correlation would predict higher CHF values under the same conditions.

6.0 References

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7.0 LIST OF ACRONYMS

1D (or 1-D)	One-dimensional
3D (or 3-D)	Three-dimensional
AECL	Atomic Energy of Canada Limited
ANS	American Nuclear Society
AP600	Advanced Passive 600 MW plant
APWR	Advanced Pressurized Water Reactor
Appendix K	Appendix K of 10 CFR Part 50
ASME	American Society of Mechanical Engineers
BLD	Blowdown period of SBLOCA
BO	Boil-off period of SBLOCA
CCFL	Countercurrent flow limited
CFR	Code of Federal Regulations
CHF	Critical heat flux
CL	Cold leg
CVCS	Chemical and Volume Control System
DCD	Design Control Document
DNB	Departure from nucleate boiling
DVI	Direct Vessel Injection
ECC	Emergency core cooling (or coolant)
ECCS	Emergency Core Cooling System
EFWS	Emergency Feedwater System
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
FRS	Fuel Rod Simulator
GDC	General Design Criteria
H2TS	Hierarchical Two-Tiered Scaling
HHIS	High-Head Injection System
HL	Hot leg
HPI	High pressure injection
ID	Inner diameter
IET	Integral effects test
INL	Idaho National Laboratory
JAERI	Japan Atomic Energy Research Institute
LBLOCA	Large break loss of coolant accident
LOCA	Loss of coolant accident
LOFT	Loss of Fluid Test
LOOP	Loss of Offsite Power
LSC	Loop seal clearance period of SBLOCA
LWR	Light water reactor
MHI	Mitsubishi Heavy Industries, LTD
MSLB	Main steam line break
NC	Natural circulation period of SBLOCA
NR	Neutron Reflector
NRC	U. S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PCT	Peak cladding temperature
PIRT	Phenomena identification and ranking table
PWR	Pressurized water reactor

RAI	Request for additional information
RCP	Reactor coolant pump
RCS	Reactor coolant system
REC	Core recovery period of SBLOCA
RG	Regulatory Guide
ROSA/LSTF	Rig of Safety Assessment/Large Scale Test Facility
RV	Reactor vessel
RWSP	Refueling Water Storage Pit
SBLOCA	Small break loss of coolant accident
SE	Safety Evaluation
SER	Safety Evaluation Report
SET	Separate effects test
SG	Steam generator
SI	Safety Injection
SRP	Standard Review Plan
THTF	Thermal Hydraulic Test Facility
TMI	Three Mile Island
UPTF	Upper Plenum Test Facility
US-APWR	United States Advanced Pressurized Water Reactor
USNRC	U. S. Nuclear Regulatory Commission