ADVANCED SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
TOPICAL REPORT MUAP-07010-P, REVISION 4
"NON-LOCA METHODOLOGY"
MITSUBISHI HEAVY INDUSTRIES, Ltd
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1.0 INTRODUCTION

By letter dated July 20, 2007, Mitsubishi Heavy Industries, LTD (MHI), the applicant, submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval of Topical Report MUAP-07010-P, Revision 0, "Non-LOCA Methodology" [Reference 1] in support of the United States - Advanced Pressurized Water Reactor (US-APWR) Design Certification (DC) Application. This report was later superseded by letter dated October 29, 2010, which transmitted Topical Report MUAP-07010-P, Revision 1 [Reference 2], by letter dated August 31, 2011, which transmitted Topical Report MUAP-07010-P, Revision 2 [Reference 3], by letter dated April 5, 2012, which transmitted Topical Report MUAP-07010-P, Revision 3 [Reference 52] and by letter dated May 11, 2012, which transmitted Topical Report MUAP-07010-P, Revision 4 [Reference 53]. Topical Report MUAP-07010-P describes the codes and the general evaluation models used to perform Non-Loss of Coolant Accident (Non-LOCA) simulations of the US-APWR reactor design to determine the consequences of such transients and accidents on the reactor core and associated piping systems. The purpose of these evaluations is to demonstrate that, for a given transient scenario, reactor fuel failures and peak system pressures on both the primary and secondary sides of the heat transfer loops remain within acceptable ranges, and to demonstrate that onsite and offsite radiological releases remain within acceptable limits. The analyses of the US-APWR transient and accident analyses are described in the US-APWR Design Control Document (DCD) [Reference 4] Tier 2 Chapter 15, "Transient and Accident Analysis."

The applicant also provided a report for the MARVEL-M computer program. MARVEL-M is described in MHI's report GEN0-LP-480, "MARVEL-M A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System." GEN0-LP-480, Revision 6, was provided in March 2009 [Reference 5]. The current version, Revision 12, was provided in October 2010 [Reference 6].

This safety evaluation (SE) documents the NRC staff's review of Topical Report MUAP-07010-P and report GEN0-LP-480. Section 2 of this SE describes the US-APWR reactor systems in the context of the Non-LOCA analysis. The regulatory basis and acceptance criteria for Standard Review Plan Chapter 15 Non-LOCA analyses are provided in Section 3. A summary of MUAP-07010-P, including the computer codes used in the Non-LOCA analysis is provided in Section 4. The NRC staff technical evaluation of Topical Report MUAP-07010-P is provided in Section 5. Discussions of requests for additional information (RAIs) issued as a part of the review are provided, along with staff evaluations of the applicant's RAI responses and the acceptability of the corresponding analytical models. Section 6 provides the staff's conclusions regarding the adequacy of the codes and methods described in Topical Report MUAP-07010-P for representing the relevant phenomena and demonstrating compliance with the regulatory acceptance criteria, as well as any conditions and limitations on the codes and methods for licensing calculations. References are listed in Section 7. The NRC staff evaluation of GEN0-LP-480 is provided in Section 5 as Appendix A.

2.0 SYSTEM DESCRIPTION

The general system configuration of the US-APWR is equivalent to that of a Westinghouse designed four-loop pressurized-water reactor (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 megawatt thermal (MWt).

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

- Primary System (reactor core, reactor vessel (RV), reactor coolant system (RCS), emergency core cooling system (ECCS))
- Secondary System (main steam system, main feedwater system (MFWS), emergency feedwater system (EFWS))
- Containment Vessel

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

2.1 Primary System Components

The primary system includes the reactor core, RV, RCS and the ECCS, which are included in the US-APWR MARVEL-M Non-LOCA model.

2.1.1 Reactor Core and Reactor Vessel

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 approximately 14-ft (4.2-m) active fuel length.

The RV internals consist of the lower and upper core support assemblies and the neutron reflector. These RV internals support the core, maintain fuel assembly and control rod alignments, limit fuel assembly movement, direct the coolant flowing through the fuel assemblies, guide the in-core instrumentation and provide RV radiation shielding.

2.1.2 Reactor Coolant System

The RCS consists of the RV, the SGs, the reactor coolant pumps (RCPs), the pressurizer, and the reactor coolant pipes and valves connecting to and interconnecting those components.

The RV, which contains the structures discussed above in Section 2.1.1, has four inlet nozzles, four outlet nozzles, and four safety injection nozzles located between the RV upper 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 design similar to Westinghouse 93A pumps used in four-loop PWRs, driven by three-phase induction motors. A flywheel on the shaft above the motor provides additional inertia to extend pump coast-down. 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 safety injection signal generated by low pressurizer pressure or high containment pressure, as required by Three Mile Island (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.

2.2 Emergency Core Cooling System

The ECCS injects borated water into the RCS following some anticipated operational occurrences and postulated accidents 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 an inadvertent opening of a steam generator relief or safety valve or 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:

- Large-break LOCA and small-break LOCA of the primary piping,

- Control rod ejection,

- inadvertent opening of a steam generator relief or safety valve,

- Main steam line break,

- Steam generator tube rupture (SGTR).

(2) The ECCS is designed with sufficient redundancy (four independent 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 core cooling following an accident with one train out of service.

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

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

The ECCS includes the accumulator system, the high-head injection system (HHIS) system, and the emergency letdown system. The accumulator system and HHIS system are included in the US-APWR Non-LOCA evaluation model.

2.2.1 Accumulator System

The accumulator system, 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 reactor vessel, and then provides a smaller injection flow during core reflooding in association with the HHIS injection pumps. The HHIS provides long term core cooling.

2.2.2 High-Head Injection System

The HHIS, which is an active safety component, consists of four independent trains, each containing a safety injection pump (SIP) and the associated valves and piping. The safety coolant is directly injected into the downcomer using direct vessel injection (DVI). The SIPs start automatically upon receipt of the SI signal. One of four independent safety electrical buses is available to each SIP. The SIPs 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 break LOCA (LBLOCA) or small break LOCA (SBLOCA). This capability ensures adequate emergency core cooling (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 RWSP, in the containment, provides a continuous borated water source for the SIPs. This configuration eliminates the need for realignment from the refueling water storage tank to the containment sump, which is employed in existing PWR plants.

2.3 Secondary System Components

The secondary system consists of the main steam system (MSS), the MFWS, the EFWS, and the power conversion system.

2.3.1 Main Steam System Components

The MSS includes the main steam lines from the SG outlets to the turbine inlet steam chests and equipment and piping connected to the main steam lines. The main steam safety and relief

valves are installed upstream of the main steam isolation valve (MSIV). They prevent excessive steam pressure and maintain cooling of the RCS if the turbine bypass is not available. The total capacity of the main steam safety valves (MSSV) exceeds 100 percent of the rated main steam flow rate. Branch pipes for driving the turbine-driven emergency feedwater (EFW) pumps are connected upstream of the MSIVs. The secondary sides of SGs are included in the US-APWR Non-LOCA MARVEL-M model, up to the turbine. The steam line relief and isolation valves, and the steam dump control valve, are included in the MARVEL-M model.

2.3.2 Main Feedwater System Components

The MFWS 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 that are installed on the feedwater lines. The feedwater control and isolation valves on the secondary sides of the SGs are included in the MARVEL-M model.

2.3.3 Emergency Feedwater System Components

The EFWS consists of two motor-driven pumps, two steam turbine-driven pumps, two EFW pits, and associated piping and valves. The four EFW pumps take suction from two EFW pits. The EFWS removes reactor decay heat and RCS residual heat through the SGs following transient conditions or postulated accidents. The EFWS is modeled as a fill system in the MARVEL-M model.

2.4 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 MARVEL-M model, an atmospheric condition inside the containment is assumed as the pressure boundary condition for systems that interact with the containment.

3.0 REGULATORY BASIS

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 34, paragraphs (a)(4) and (b)(4) specifies that each boiling or pressurized light-water nuclear power reactor must provide the following:

> ... analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The NRC has provided guidance regarding how the above regulatory criteria can be met. Regulatory Guide (RG) 1.203 [Reference 7] describes acceptable approaches to develop the computer programs to conduct transient and accident analyses. RG 1.206 [Reference 8], Section 3.I.15 provides guidance for the evaluation of transients and accidents. Section 15 of the standard review plan (SRP) [Reference 9] discusses specific acceptance criteria for each transient.

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

1. 10 CFR Part 20, “Standards for Protection Against Radiation”

2. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities” (especially 10 CFR 50.46 and Appendix A)

3. 10 CFR Part 100, “Reactor Site Criteria”

4. 10 CFR part 52, “Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants”

The following General Design Criteria (GDC) from Appendix A to 10 CFR 50 are relevant to Section 15 of SRP:

1. GDC 4 as it relates to the requirement that structures, systems, and components (SSC) important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including such effects as pipe whip and jet impingement.

2. GDC 10, as it relates to the reactor coolant system (RCS) being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDL) are not exceeded during normal operations including anticipated operational occurrences (AOO).

3. GDC 13, as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operation, for AOOs, and for accident conditions.

4. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

5. GDC 17, as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety.

6. GDC 20, as it relates to the reactor protection system (RPS) being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed SAFDLs during any condition of normal operation, including AOOs.

7. GDC 25, as it relates to the requirement that the RPS be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.

8. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs.

9. GDC 27 and 28, as they relate to the reactivity control system being designed with appropriate margin to ensure that SAFDLs are not exceeded and that the capability to cool the core is maintained.

10. GDC 29, as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs.

11. GDC 34, as it relates to the capability to transfer heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.

12. GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.

13. GDC 55, as it relates to the isolation requirements of small-diameter lines connected to the primary system.

SRP 15.0, “Introduction – Transient and Accident Analyses.” defines anticipated operational occurrences (AOOs) and postulated accidents (PAs) as the following:

> AOOs, as defined in Appendix A to 10 CFR Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.
>
> PAs are unanticipated occurrences (i.e., they are postulated but are not expected to occur during the life of the nuclear power plant).

The acceptance criteria for the safety analyses of the AOOs and PAs for compliance with the above relevant regulations and GDCs are specified in SRP Section 15.0 as described below.

The following are the acceptance criteria specified in SRP 15.0 necessary to meet the requirements GDCs for AOOs:

i. Pressure in the reactor coolant (P_{RCS}) and main steam (P_{MS}) systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

ii. Fuel cladding integrity is maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

iii. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

GDC 10 within Appendix A to 10 CFR 50 establishes that SAFDLs should not be exceeded during any condition of normal operation, including the effects of AOOs.

The following are the acceptance criteria specified in SRP 15.0 necessary to meet the requirements of the GDCs for PAs:

i. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits.

ii. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit for PWRs. If the minimum DNBR does not meet this limit, then the fuel is assumed to have failed.

iii. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.

iv. The postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For the Reactivity Initiated Accidents (RIAs), SRP 4.2 Appendix B provides the following additional fuel cladding failure acceptance criteria:

> The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/gm (306 Btu/lbm) for fuel rods with an internal rod pressure at or below system pressure and 150 cal/gm (270 Btu/lbm) for fuel rods with an internal rod pressure exceeding system pressure. For intermediate (greater than 5 percent rated thermal power) and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR).
>
> The pellet cladding mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit shown in Figure B-1 of SRP 4.2 Appendix B.

Appendix B of SRP 4.2 also specifies the following additional acceptance criteria regarding core coolability

1. Peak radial average fuel enthalpy must remain below 230 cal/gm (414 Btu/lbm).

2. Peak fuel temperature must remain below incipient fuel melting conditions.

3. Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to pressure boundary, reactor internals, and fuel assembly structural integrity.

4. No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

4.0 SUMMARY OF MUAP-07010-P

In Section 1.0 of the subject topical report the applicant provided a high-level overview of the codes and methods used for US-APWR Non-LOCA safety analysis, the pedigree of each code, and an outline for the rest of the report.

Section 2.0 of the topical report provided specific details regarding the methods embedded in each of the computer codes employed for US-APWR Non-LOCA evaluations. It contains descriptions of modifications that MHI made to Westinghouse Electric Corporation (WEC) codes MARVEL and TWINKLE. All changes have been made under MHI's formal Quality Assurance Program (QAP), documented in [Reference 16].

For Non-LOCA safety analysis, MHI uses MARVEL-M [Reference 6] to calculate the plant system response to transient conditions, TWINKLE-M [Reference 11] to evaluate spatial neutron kinetics for those transients when necessary, and VIPRE-01M [Reference12] to calculate detailed core subchannel thermal-hydraulics and fuel temperature transients. Both MARVEL-M and TWINKLE-M are applicant modified versions of WEC codes MARVEL and TWINKLE. MARVEL-M is based on an early version of MARVEL which has not received NRC approval (see RAI 2.1-21 [Ref. 20], response for details). TWINKLE-M is based on the NRC approved version of TWINKLE [Reference 14]. Section 2.1.3, "Theoretical Models of MARVEL-M Improvement" of MUAP-07010 describes improvements to the MARVEL-M theoretical model. The MARVEL-M code is the same as the original MARVEL code from the perspective of the constitutive and principal models. The main differences between the original MARVEL code and MARVEL-M are the extension from two-loop simulation to four-loop simulation and the addition of a built-in RCP model. The other refinements include a pressurizer surge line node, a hot spot heat flux simulation model, and improved numerical solution and conversion techniques.

VIPRE-01M is the MHI-specific version of the VIPRE-01 [Reference 15] subchannel thermal-hydraulics code developed by Battelle Pacific Northwest Laboratory under Electric Power Research Institute (EPRI) sponsorship. VIPRE-01 has been approved for use in the United States for pressurized water reactor licensing analysis.

The staff notes that the advanced nodal code (ANC) core simulation code [Reference 17] is not specifically identified in topical report Section 2.0 as a code used for the Non-LOCA analyses. However, Sections 3, 5, and 6 of the topical report describe use of the ANC code and results in the Non-LOCA methodology as a basis for models, the validation and assessment of models, and the checking and generating of core reactivities and power distributions.

Section 3.0 of the topical report provided the code validation evaluation performed by MHI for MARVEL-M and TWINKLE-M.

To validate the adequacy of the modifications included in the MARVEL-M code, MHI performed analyses of four events (Uncontrolled rod cluster control assembly (RCCA) Bank Withdrawal at Power, Complete Loss of Forced Reactor Coolant Flow, Partial Loss of Forced Reactor Coolant Flow, and Reactor Coolant Pump Shaft Seizure) using the MARVEL-M code and the four-loop LOFTRAN [Reference 18], and compared the calculated results.

The solution methods and constitutive models of the TWINKLE-M code have not changed from the original version, but the maximum number of spatial mesh points was expanded from 2,000 points to a variable number input by the user.

The three-dimensional calculation using the TWINKLE-M code was verified by comparing the power distribution with that from the core simulator ANC code [Reference 17]. For three-dimensional transient analyses, it is desirable to use as coarse a mesh as possible while maintaining sufficient accuracy. A second objective of the validation was to compare the results of a two-by-two coarse mesh simulation for the rod ejection accident to a four-by-four fine mesh simulation for the same accident with the same cross-section data using TWINKLE-M for both. Cases with and without an ejected rod under steady-state conditions were analyzed. Sensitivity studies were also performed to compare the TWINKLE-M results for different mesh size assumptions.

Section 4.0 of the topical report provided information on event classifications and acceptance criteria for SRP Chapter 15 Non-LOCA events. The US-APWR analyses for Non-LOCA events are grouped into six categories. The topical report lists the events within each category, their classifications, the computer codes employed in the analysis and the event-specific acceptance criteria. The staff reviewed the acceptance criteria for the design basis events specified in Tables 4.2-1, “Events in Increase in Heat Removal from the Primary System,” 4.3-1, “Events in Decrease in Heat Removal by the Secondary System,” 4.4-1, “Events in Decrease in Reactor Coolant System Flow Rate,” 4.5-1, “Events in Reactivity and Power Distribution Anomalies,” 4.6-1, “Events in Increase in Reactor Coolant Inventory,” and 4.7-1, “Events in Decrease in Reactor Coolant Inventory,” and found them consistent with the regulatory acceptance criteria.

Section 5.0 of the topical report provided a description of the event-specific methodology. Depending on the nature and computational capabilities needed for adequate simulation of specific transients and accidents, the methodology employs a single computer code, or a combination of computer codes, such that the analyses fall into one of the following three categories:

- Analyzed using MARVEL-M only.
- Analyzed using MARVEL-M and VIPRE-01M in sequence.
- Analyzed using TWINKLE-M and VIPRE-01M in sequence.

The first category that uses MARVEL-M alone includes most of the Non-LOCA transients that challenge the design limits for the RCS and main steam system pressure limits, as well as loop-symmetric accidents at full-flow conditions that fall within the capabilities of the simplified MARVEL-M DNBR model. These accidents do not require detailed calculation of localized fuel parameters and do not require spatially dependent transient calculations for accident-specific power levels or power distributions.

The second category that uses MARVEL-M in combination with the VIPRE-01M subchannel thermal-hydraulic code is used for accidents that challenge the DNB design limits under reduced flow conditions such as the partial loss of flow, complete loss of flow, locked RCP rotor, or RCP sheared shaft conditions. The loop-dependent and core total flow, core inlet conditions, pressure, and power are calculated using the MARVEL-M program, and then the VIPRE-01M code is used to determine the hot channel or hot spot fuel response including DNBR, fuel temperatures, and cladding temperature.

The third category that uses TWINKLE-M in combination with the VIPRE-01M code is reserved for rapid reactivity transients requiring space- and time-dependent nuclear power and power distribution calculations for input to a detailed fuel response calculation. The evaluation of the Spectrum of RCCA Ejection event was provided as an example, including the 3-D TWINKLE-M capabilities as needed.

Section 6.0 of the topical report provided the sample transient analysis results for six events:

- Uncontrolled RCCA Bank Withdrawal at Power.
- Complete Loss of Forced Reactor Coolant Flow.
- Spectrum of RCCA Ejection.
- Steam System Piping Failure.
- Feedwater System Pipe Break.
- SGTR.

Section 7.0 of the topical report provided the applicant's conclusion that the existing codes and methodologies are appropriate for US-APWR analyses.

5.0 TECHNICAL EVALUATION

5.1 Code Descriptions and Validations

Section 2 of MUAP-07010-P(R4), "Non-LOCA Methodology" [Reference 53] describes MARVEL-M, TWINKLE-M and VIPRE-01M; and Section 3 provides MARVEL-M and TWINKLE-M code validation. The staff review of Section 2 (computer code description) and Section 3 (code validation) of that report is presented here.

5.1.1 MARVEL-M Code

In Section 2.1 of the Non-LOCA methodology topical report, the applicant discussed the history of MARVEL-M development, an overview of the MARVEL-M code, and the mathematical model improvements in the MARVEL-M code. In support of the Non-LOCA methodology topical report, the applicant also submitted MARVEL-M code manual report GEN0-LP-480, "MARVEL-M, A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System," (References 5 and 6), which provides detailed descriptions of various models in MARVEL-M.

The MARVEL-M code simulates the reactor core, reactor vessel, reactor coolant loops, SGs, control and protection systems, and safety injection systems of two-, three-, and four-loop PWRs. Topical report Section 2.1.2, "General Description – Overview," gives an overview of various models in MARVEL-M. The details of these models are described in the MARVEL-M manual GEN0-LP-480. Therefore, the staff evaluation of MARVEL-M is based on the detailed description in the MARVEL-M manual.

The MARVEL-M code was developed by modifying the original two-loop MARVEL code [Reference 13] to expand its capability for four-loop simulation. Topical report Section 2.1.3, "Theoretical Models of MARVEL-M Improvement," describes the mathematical model improvements in the MARVEL-M code. The improvements include the expansion to the four-loop reactor coolant system model; four-loop flow mixing model in the reactor vessel; reactor coolant pump and flow transient mode; secondary steam system model; and other model refinements in the pressurizer surge line model, hot-spot fuel kinetics model, core void simulation, feedline break blowdown simulation, and conversion of RCS volume balance by pressure search.

It was noted that the applicant did not base the development of MARVEL-M on the approved version of MARVEL, but rather on an earlier version. RAI 2.1-21 therefore asked the applicant to detail the difference between MARVEL-M and the approved version of MARVEL. In its response to RAI 2.1-21 [Ref. 20], the applicant confirmed that the development of MARVEL-M was based on a version which preceded the approved version of MARVEL and that it was developed independently of Westinghouse. The applicant identified the additional features in the approved version of MARVEL from the base MARVEL version which MHI used to develop MARVEL-M. The applicant first incorporated equivalent models for these additional features into the base version of MARVEL, and verified them by comparison of MARVEL-M to the LOFTRAN [Ref. 18] simulations. The applicant then incorporated its own specific modifications and improvements into the final version of MARVEL-M. This response clearly describes the lineage of MARVEL-M and helped focus the staff's review. However, because of extensive changes from the MARVEL code, the staff conducted a detailed review of the MARVEL-M code described in GEN0-LP-480 and the methods used for verification of the modified code. The detailed staff review of MARVEL-M is described in Appendix A of this report.

This section summarizes the staff's evaluation of major model improvements and refinements described in Section 2.1.3 of the topical report.

Four-Loop Reactor Coolant System Model:

One major MARVEL-M improvement over MARVEL, described in Section 2.1.3.1 of the topical report, is the ability to simulate up to four reactor coolant loops. Although the hydraulic and thermal models of the individual reactor coolant flow sections and the model of the SGs and pressurizer are the same as the original MARVEL two-loop models, the MARVEL-M RCS flow nodalization has been expanded to up to four reactor coolant loops that can be simulated by the code, and the algorithms for core mixing in the reactor vessel and the steam lines, respectively, have changed to accommodate the expansion of the number of coolant loops.

The original MARVEL code used six different flow modules to describe the thermal behavior of the RV and coolant loops: a HEAT model for the core heated section; a MIXG module for mixing plenums including the RV outlet plenum, SG inlet and outlet headers, RV inlet plenum, and reactor core bypass; a MIXD module to evaluate the thermal-hydraulic behavior in the reactor vessel upper head dead volume; a MIXS module for the upper plenum region; a SLUG module to model the reactor coolant system pipes where the fluid flow is such that exit fluid properties are not affected by inlet fluid properties; and a HEEX module for the primary side SG tubes. MARVEL-M maintains the same flow modules. Section 1.3 of GEN0-LP-480 provides detailed descriptions of these flow modules. The staff's review of these flow modules is described in Section A.3 of Appendix A of this report. Figure 2.1-3, "Reactor Coolant System Flow Model," of the topical report shows the MARVEL-M nodal representation of the four-loop reactor coolant system. This flow model is the same as the MARVEL model except that the RCS flow loop is increased to four loops, the downcomer and lower plenum include four MIXG flow channels each, and the reactor core is expanded to four HEAT flow channels.

In RAI 2.1-3, the staff requested the applicant provide a MARVEL/MARVEL-M code comparison for a typical two-loop configuration calculation. In its response to RAI 2.1-3 [Ref. 19], the applicant provided detailed comparisons between the MARVEL and MARVEL-M simulations of the "Loss of Load" (a uniform event) and the "Feedwater Pipe Break" (a non-uniform event). The MARVEL-M calculations modeled all four loops separately, while MARVEL modeled one loop separately and the other three loops lumped together. The results for the two code simulations were essentially identical. These comparisons demonstrate that the MARVEL and MARVEL-M predictions are very similar and therefore provide reasonable assurance that the four-loop modeling is correctly implemented in MARVEL-M.

Flow Mixing in Reactor Vessel (Four-Loop Model):

The MARVEL-M algorithm for flow mixing in the RV was changed from the original MARVEL model as described in topical report Section 2.1.3.2. The changes to the flow mixing algorithm were made to accommodate the expansion of the modeling capability from two to four loops. There is no fundamental change in the mixing phenomenology, and the basic assumptions are the same as in the two-loop MARVEL version. The reactor coolant fluid circulating in the reactor coolant loops is introduced into the reactor vessel through the inlet nozzles (cold legs). Thus the mixing in the RV inlet and outlet plenum is imperfect. In order to take this into consideration in the analysis of the RV thermal kinetics behavior, an azimuthal as well as an axial analysis is necessary. The azimuthal effect is considered by using a maximum of four separate flow channels for each loop simulated.

The thermal and hydraulic behaviors are simulated by simple mixing models using representative or conservative user inputs of the mixing factors for the mixing in the reactor vessel downcomer/lower plenum and upper plenum, respectively. The input mixing factors can be derived from the 1/7 scale mixing tests as described in Topical Report MUAP-07022-P, "US-APWR Reactor Vessel Lower plenum 1/7 Scale Model Flow Test Report," [Ref. 23]. As discussed in Appendix A, Section A.4 of this report, the staff evaluated and found the flow mixing algorithm acceptable.

Secondary Steam Generator Model (Four-Loop Model):

Topical report Section 2.1.3.4 describes the improvements of the MARVEL-M SG and secondary system model for expansion to four-loop capability. Figure 2.1-6, "Steam Line Model," shows the main steam lines from each SG are connected together at a common steam header, each via an isolation valve and a check valve. If the operating conditions of the SGs are different from each other, the steam outputs from the SGs are unbalanced. The steam flow distribution is then dependent upon the steam pressure of each SG and upon the pressure losses through the steam lines. A detailed description of the secondary steam generator model is provided in Section 1.8 of GEN0-LP-480. As described in Section A.8 of Appendix A of this report, the staff's evaluation of the secondary SG model and the method of evaluating the steam flow distribution found it acceptable.

Reactor Coolant Pump and Flow Transient Model:

The original MARVEL code used a simplified empirical flow coastdown model and a transition from this flow model to natural circulation. MARVEL-M uses an explicit RCP model. Topical report Section 2.1.3.3 describes the RCP and flow transient models. The fundamental flow transient equations are based on a momentum balance around each reactor coolant loop and across the RV, flow continuity, and RCP characteristics with or without electrical power supply. The RCP head is derived from the homologous curve of the RCP, which depends on the pump speed. When the electric power supply to an RCP is lost, the pump motor torque is lost and flow coastdown results. The pump head decreases according to the decrease in pump speed and is eventually lost, which results in natural circulation. MARVEL-M includes the RCP hydraulic kinetic model, flow coastdown model, and natural circulation elevation head model. The detailed description of the transient RCS flow performance is provided in Section 1.7 of GEN0-LP-480. As described in Section A.7 of Appendix A of this report, the staff found the RCP and RCS flow transient model acceptable.

Pressurizer Surge Line Model:

As shown in Figure 2.1-3, "Reactor Coolant System Flow Model," in Section 2.1.3.1 of the topical report, MARVEL-M adds a flow section in the pressurizer surge line between the hot leg connection and the pressurizer. The addition of the surge line section in the RCS flow model is a MARVEL-M refinement to more realistically model pressurizer insurge enthalpy. If the surge line is not simulated, hot leg coolant water directly enters the pressurizer during insurge, which may result in over-predicting cooling of the pressurizer liquid phase and may cause a larger pressurizer pressure reduction for a subsequent outsurge. This refinement resulted from the observation of a transient test during a reactor plant preoperational test. The staff agrees that explicitly modeling the pressurizer surge line rather than lumping it with the pressurizer will result in more accurately representing the hydraulic behavior occurring during pressurizer in-surge and out-surge. Therefore, the staff finds this MARVEL-M modeling refinement acceptable.

The MARVEL-M core and RCS thermal-hydraulic model does not simulate the pressure gradient in the system except for the pressurizer and the RCS at the hot-leg pressurizer surge line connection. The reactor coolant flow is treated as a homogeneous equilibrium mixture. The staff questioned what model MARVEL-M uses for two-phase flow and how the user deals with conditions in which homogeneous two-phase flow is not applicable. In its response to RAI 2.1-4 [Ref. 19], the applicant noted that, for the majority of Non-LOCA events, voids exist only in the pressurizer and RV upper head region, and that boiling beyond homogeneous two-phase flow does not occur for Non-LOCA conditions. Furthermore, MARVEL-M provides a message to the analyst if boiling occurs, warning the analyst to review the results to evaluate the significance of the voiding on the calculated results. This is acceptable because MARVEL-M will not be applied to events exhibiting non-homogeneous flow.

Hot-Spot Fuel Thermal Kinetics Model:

The hot-spot fuel thermal kinetics model in the original MARVEL code was similar to the fuel thermal kinetics model for the average channel. MARVEL-M includes a more detailed fuel thermal kinetics model adopted from the Westinghouse FACTRAN code [Ref. 51], which had been approved by the NRC. The FACTRAN code has the ability to model up to 10 radial sections in the fuel pellet, cladding and clad surface to compute the transient fuel temperature and heat flux. The MARVEL-M hot spot fuel model is for the computation of the heat flux transients at the surface of the cladding at the hot spot. The normalized hot-spot heat flux can be used as an option (the largest heat flux between the average channel and hot spot is used) to calculate DNBR using the simplified DNBR model in MARVEL-M. The fuel pellet thermal properties can be input by the user. This model is described in Section 1.2.3 of GEN0-LP-480. As described in Section A.2.3 of Appendix A of this report, the staff's evaluation finds this model acceptable.

Core Void Simulation:

MARVEL-M has a model to calculate the boiling of coolant or void fraction in the core when the core power increases enough to cause voiding or if the core coolant temperature exceeds the saturation temperature. However, this core thermal-hydraulic model is not sufficiently detailed to compute the void formation at accidents such as rod ejection that could cause locally high void fraction in a high power region in the core. Rapid excessive void formation in the core results in an in-surge to the pressurizer and a pressure increase. To adequately predict the pressure increase, the MARVEL-M model includes a feature scheme to accept void transients calculated by an external, detailed thermal-hydraulic code, such as VIPRE-01M [Ref. 12], which can compute void formation taking into account subcooled boiling, detached boiling, and bulk boiling. This feature is only used to assure that the RCS pressure is conservatively high for the rod ejection accident where local voids in the core could impact the peak pressure.

As stated in Section A.15 of Appendix A of this report, the safety analysis of the rod ejection event uses the VIPRE-01M code to calculate the void transient data input to MARVEL-M for the purpose of calculating the peak RCS pressure. The staff finds this core void simulation feature acceptable as VIPRE-01M has the detailed thermal-hydraulic modes to determine the core void fraction (See the RAI 4.1-3 response and staff review described in Section A.15 of this SE).

Feedline Break Blowdown Simulation:

During a feedwater line break with water release, when the SG is rapidly depressurized below the feedwater saturation pressure, feedwater contained in the feedline flashes and a mixture of steam and water can be released into the SG shell side.

MARVEL-M includes a feedline blowdown simulation model to simulate this phenomenon. Each feedwater line is assumed to contain fluid with the enthalpy of the feedwater when the feedwater line isolation valves are closed. After the individual SG pressure falls below the saturation pressure of the feedwater enthalpy, water is assumed to push into the SG by void formation. This feedwater flashing model is described in Section 1.5.5 of GEN0-LP-480. The applicant states that this feedwater flashing model is only used in the DCD Chapter 6, "Engineered Safety Features," mass and energy release analysis. As described in Section A.5.5 of Appendix A of this report, the staff did not perform an evaluation since this model is not used for Chapter 15, "Transient and Accident Analyses," analysis.

Conversion of RCS Volume Balance by Pressure Search:

[(Proprietary information withheld under 10 CFR 2.390)

] This method is also described in Section 4.0(1) of GEN0-LP-480. The staff's evaluation, described in Section A.15 of Appendix A of this report, finds this method acceptable.

Realistic Models

MARVEL-M contains "Realistic Models" as options for simulation of real plant transient behavior, as discussed in Section 2.1.4 of the topical report. The applicant stated that these models are code options and are not used for licensing evaluations of reactor plants. Therefore, the staff does not evaluate the realistic models.

Limitations for MARVEL-M Application

MUAP-07010, Section 2.1.5.1 summarizes the limitations of range of operating variables for the application of MARVEL-M in terms of the RCS temperature and pressure, pressurizer water level, SG steam pressure and water inventory, reactor coolant loop flow, and reactor core kinetics. These limitations are consistent with the approved version of MARVEL. Topical report Section 2.1.5.2 discusses the applicability of MARVEL-M to the scenarios of licensing analysis. Since MARVEL-M uses space independent point neutron kinetics, the reactivity initiated events, such as an inadvertent RCCA withdrawal from subcritical condition, should be analyzed with

other codes such as TWINKLE-M which is a 1 or 3 dimensional model to capture spacial effects. Since MARVEL-M uses a simplified DNBR calculation, those transients, such as a loss of flow for which the minimum DNBR is heavily dependent on changes in reactor coolant flow, should utilize other subchannel codes such as VIPRE-01M for the DNBR calculation. Also since MARVEL-M assumes homogeneous equilibrium two-phase flow for the RCS flow, it should not be used for LOCA analysis.

In addition, as described in Section A.6.3 of Appendix A of this report, since MARVEL-M has not been qualified for simulating the discharge of liquid through the pressurizer safety and relief valves, the usage of MARVEL-M is restricted to events which only discharge steam through the pressurizer safety and relief valves.

Topical report Section 4 describes the computer codes used for each of the Chapter 15 Non-LOCA design basis events. The original MARVEL was approved for use to analyze only four events: (1) steam line rupture, (2) feedwater line rupture, (3) startup of an inactive reactor coolant loop, and (4) excessive heat removal due to a feedwater system malfunction. MARVEL-M is used to analyze many more events than these four events. In its response to RAI 2.1-23 [Ref. 20], regarding the acceptability of MARVEL-M for the Non-LOCA events not listed above the applicant noted that while the MARVEL code was not approved for analyzing other Non-LOCA events, it still produced acceptable results for those events. The applicant noted that the disturbances caused by many Non-LOCA events were milder than those events for which MARVEL was approved. In its responses to RAI 2.1-17, and RAIs 3.1-2 through 3.1-5 [Ref. 19], the applicant also noted that LOFTRAN has been approved by the NRC for Non-LOCA event analyses and that the comparisons of the MARVEL-M results to LOFTRAN results were good as shown in Sections 3.1.1 through 3.1.4 of the topical report. MARVEL-M results have also been favorably compared to results from the original two-loop MARVEL (response to RAI 2.1-3 [Ref. 19]), and to the measured data from a four-loop plant for a partial loss of forced reactor coolant flow and a complete loss of forced reactor coolant flow (response to RAI 2.1-16 [Ref. 21]). These comparisons demonstrated that MARVEL-M was acceptable for DCD Tier 2 Chapter 15 Non-LOCA analyses. Based on its review of the RAI responses (i.e., MARVEL-M compares well with LOFTRAN and plant data), and the MARVEL-M validation as described in Section 5.1.2 of this report, the staff agrees with the applicant's conclusion that MARVEL-M is acceptable for use in Non-LOCA events.

5.1.2 MARVEL-M Code Validation

Section 3.1 of the topical report discusses the validation of the MARVEL-M code. The validation of MARVEL-M for US-APWR Non-LOCA analysis consisted of comparing calculated results to those of the four-loop LOFTRAN code (Ref. 18). The applicant stated that a code-to-code comparison is sufficient for validation because LOFTRAN had been used extensively in the licensing analysis of currently operating nuclear plants in the United States for the events that are analyzed with MARVEL-M. The applicant presented code comparisons for four events: uncontrolled RCCA bank withdrawal at power, partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, and reactor coolant pump shaft seizure. Based on the information provided by the applicant, the staff agrees that MARVEL-M and LOFTRAN are in agreement but the Staff issued follow-on RAIs regarding the uncontrolled RCCA bank withdrawal event.

Topical report Section 3.1.1 describes the comparison between the MARVEL-M and LOFTRAN analysis results of the uncontrolled RCCA bank withdrawal at power event. The staff requested the applicant to provide documentation of MARVEL-M/LOFTRAN code comparisons that may

not be in agreement, along with any explanations for the deviations. In its response to RAI 3.1-2 [Ref. 19], the applicant presented the results of an Uncontrolled RCCA Bank Withdrawal at Power event with an active pressure control system. For this case there were small differences in the responses for reactor power, core heat flux, average RCS temperature and pressurizer pressure between the MARVEL-M and LOFTRAN results. The applicant explained that the differences could be attributed to the following differences between the spray models in LOFTRAN and MARVEL-M:

MARVEL-M
- Saturated steam is assumed in the pressurizer.
- If cold water is sprayed, the saturated steam quickly condenses.

LOFTRAN
- Superheated steam is assumed in the pressurizer.
- If cold water is sprayed, the spray first removes the superheat maintaining pressurizer pressure (the steam phase does not condense but shrinks).
- After sufficient water has been sprayed the pressurizer reaches a saturated condition and condensation of the saturated steam begins.

Follow-up RAI 3.1-2-1 asked if the Uncontrolled RCCA Bank Withdrawal at Power event with an active pressure control system exhibited the largest differences between LOFTRAN and MARVEL-M. In its response [Ref. 22], the applicant noted that only cases which resulted in significant differences between the results of MARVEL-M and LOFTRAN were investigated further to determine reasons for the differences. Staff's review of the uncontrolled RCCA bank withdrawal at power, partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, and reactor coolant pump shaft seizure transient cases found that the uncontrolled RCCA bank withdrawal case showed the largest differences in core power, average RCS temperature and pressure. The differences in these predicted plant parameters were very small and within typical code uncertainties.

The responses to RAI 3.1-2 and RAI 3.1-2-1 are acceptable because they demonstrate that the MARVEL-M simulations of US-APWR DCD Tier 2 Chapter 15 events compare well with LOFTRAN, which is NRC-approved for simulating such events. The applicant has adequately explained the source of differences in the two code simulations for the RCCA bank withdrawal event. However, the staff does not agree that good agreement between LOFTRAN and MARVEL-M is, by itself, a sufficient basis for approving MARVEL-M.

Topical report Sections 3.1.2, 3.1.3 and 3.1.4, respectively, provide comparisons between the MARVEL-M and LOFTRAN analysis results of a partial loss of forced RCS flow, complete loss of forced RCS flow, and RCP shaft seizure. The results show that the reactor powers, core heat fluxes, loop flow rates, and pressurizer pressures calculated by the two codes are in close agreement. The staff requested the applicant to provide the DNBR versus time predicted by both MARVEL-M and LOFTRAN for the three events analyzed for code validation. In its responses to RAIs 3.1-3, 3.1-4 and 3.1-5 [Ref. 19], the applicant explained that DNBR is calculated by the VIPRE-01M code [Ref. 12] for these events using power and flow calculated by either MARVEL-M or LOFTRAN. Figure 3.1-3.1, "Partial Loss of Forced Reactor Coolant Flow Comparison between MARVEL-M and LOFTRAN," Figure 3.1-4.1, "Complete Loss of Forced Reactor Coolant Flow Comparison between MARVEL-M and LOFTRAN," and Figure 3.1-5.1, "Reactor Coolant Pump Shaft Seizure Comparison between MARVEL-M and LOFTRAN" were provided to demonstrate that the VIPRE-01M calculated DNBRs were nearly identical whether MARVEL-M or LOFTRAN boundary conditions were used. This is acceptable

because the resultant DNBRs were nearly the same for each of the systems codes used to provide boundary conditions to VIPRE-01M.

The applicant used comparisons to LOFTRAN as one of the means to validate MARVEL-M. Such comparisons are only meaningful if the algorithms, numerical methods and correlations in the two codes are different, or developed independently. The staff asked if MARVEL-M shares any significant algorithms, numerical methods, or correlations with the version of LOFTRAN used for the comparison. In its response to RAI 3.1-8 [Ref. 22], the applicant stated that MARVEL-M and LOFTRAN are completely different codes with respect to the FORTRAN programming and that the two codes were developed independently by different individuals. While there are many features of MARVEL-M and LOFTRAN that were developed independently, a comparison of the documentation for MARVEL-M and for LOFTRAN shows that the underlying numerical algorithms of the two codes are very similar. Therefore comparison of MARVEL-M to LOFTRAN was not considered by the staff as a complete validation of MARVEL-M. However, the two codes show good agreement and LOFTRAN has been validated [Ref. 24] against experimental data.

The topical report does not provide an SGTR analysis for the MARVEL-M validation. RAI 3.1-6 requested the applicant to present MARVEL-M validation data for the SGTR event. [

(Proprietary information withheld under 10 CFR 2.390)

] The response to RAI 3.1-6 is acceptable because it demonstrates the ability of MARVEL-M to calculate the phenomena associated with an SGTR event. However, the staff's acceptance of MARVEL-M for analyzing the SGTR event is predominately based on the independent comparisons of MARVEL-M and the staff's RELAP5/MOD3.3 SGTR simulations described in Section 5.3.1 of this report and in the Technical Report ISL-NSAO-TR-10-01, "US-APWR NON-LOCA RELAP5/MOD3.3 Confirmatory Runs," [Ref. 25].

In summary, the staff evaluated the MARVEL-M code validation provided in Section 3.1 of the topical report. The applicant responses satisfactorily resolved all of the staff's concerns addressed in the RAIs. The combination of the applicant's MARVEL-M/LOFTRAN comparisons, comparison to plant data and the staff's independent RELAP5/MARVEL-M comparisons provide reasonable assurance that MARVEL-M is acceptable for analyzing the US-APWR Non-LOCA transients.

5.1.3 TWINKLE-M

The applicant described TWINKLE-M in Section 2.2 and the validation of TWINKLE-M in Section 3.2 of the topical report.

TWINKLE-M [Ref. 11], the applicant's version of TWINKLE [Ref. 14], is a multi-dimensional spatial neutron kinetics code which solves two-group transient diffusion equations using a finite-difference technique. The code is used to predict the kinetic behavior of a reactor for those transients that cause a major perturbation in spatial neutron flux, specifically rod ejection accidents and uncontrolled control rod assembly withdrawal from subcritical.

The fundamental basis for the TWINKLE-M reactor kinetics code was the TWINKLE code developed by the Westinghouse Electric Corporation and approved for use to calculate the neutronics aspect of reactivity transients by the Atomic Energy Commission in 1975 [Ref. 14]. TWINKLE was a multi-dimensional neutron kinetics analysis code. TWINKLE-M employed a time-dependent hot channel factor based on the three-dimensional kinetics model in the TWINKLE code for the analysis of RCCA ejection from the hot zero power condition. The solution method and constitutive models of TWINKLE-M have not changed from the base TWINKLE code, but the maximum number of spatial mesh points is expanded from 2000 points to a variable number input by the user.

Since TWINKLE-M evolves from the NRC-approved TWINKLE code, the staff requested that the applicant list and describe all the differences between TWINKLE-M and TWINKLE; and elaborate further on the development history of TWINKLE-M. In its responses to RAIs 2.2-1 and 2.2-2 [Ref.19 and 20] respectively, the applicant presented Figure 2.2-2.1, "Historical Development of the TWINKLE Code" showing the historical relationship of TWINKLE and TWINKLE-M and stating that the solution methods and constitutive models in the two codes were identical. However, TWINKLE-M has an expanded spatial mesh to support three-dimensional core calculations and uses a discontinuity factor to improve the representation of the local power distribution in three-dimensional calculations. All other differences were confined to the treatment of input and output.

The fuel effective temperature model used to calculate Doppler feedback is of particular importance because it limits the reactor power increase caused by a reactivity insertion. In RAI REA-3, the staff requested that the applicant verify that the effective fuel temperature model used in TWINKLE-M was identical to that used in the original TWINKLE code, and to justify its applicability in light of other effective fuel temperature models developed since TWINKLE was approved (for example, the model used in the OECD PWR REA Benchmark experiment documented in NEACRP-L-335 [Ref. 26]).

In its response to RAI REA-3 [Ref. 27], the applicant provided a description of the effective fuel temperature model used in TWINKLE-M, and compared it with the model specified in NEACRP-L-335 [Ref. 26]. Using VIPRE-01M calculated radial fuel temperature distributions, the applicant demonstrated that the difference between the effective fuel temperature using the TWINKLE-M model [(Proprietary information withheld under 10 CFR 2.390)]. The applicant also provided sensitivity calculations using both effective fuel temperature models for US-APWR rod ejection transients at Beginning of Cycle (BOC) Hot Full Power (HFP) conditions and End of Cycle (EOC) Hot Zero Power (HZP) conditions. This sensitivity analysis demonstrated that there is negligible difference in the calculated maximum fuel centerline temperature, fuel average temperature, and peak reactor power for each case. Finally, the applicant provided the results

of pin-level Monte Carlo calculations that indicated the use of the TWINKLE-M model for effective fuel temperature versus using a fine-node temperature distribution from VIPRE-01M resulted [(Proprietary information withheld under 10 CFR 2.390)] conditions. Based on these calculations, and noting the empirical nature of any effective fuel temperature calculation methodology, the applicant states that the model contained within TWINKLE-M is sufficient. This is reinforced by the conservative 20 percent reduction in Doppler feedback employed within the applicant's safety analysis methodology. The staff agrees that the effective fuel temperature model in TWINKLE-M is acceptable as the applicant has demonstrated that it is similar to TWINKLE and has negligible effect on Chapter 15 analysis results.

In light of the calculations presented by the applicant, the staff finds the methods employed by the applicant to calculate the effective fuel temperature for Doppler feedback to be acceptable, especially in concert with the conservative 20 percent reduction in Doppler feedback.

Another model embedded within TWINKLE-M the staff reviewed, based upon its ranking in the phenomena identification and ranking table (PIRT) [Ref. 28], was the fuel specific heat capacity. This particular material property received an importance ranking of "High" in an NRC-sponsored PIRT [Ref. 28] due to its importance in determining the transient fuel temperature during a rod ejection event, which impacts the calculated Doppler feedback and the magnitude of the power excursion. The TWINKLE topical report [Ref. 14] describes a simple linear model for fuel specific heat capacity that was developed in the early 1970's. More accurate models have been developed in subsequent years and published in the Material Properties database [Ref. 29]. No reference to this updated model was found in the applicant's responses to RAIs requesting that the applicant detail changes made from TWINKLE to TWINKLE-M; thus the staff issued RAI REA-13 requesting that the applicant justify continued use of the original TWINKLE model.

In its response to RAI REA-13 [Ref. 30], the applicant explained that in the process of updating the fuel property models in TWINKLE-M, the MATPRO model for fuel specific heat capacity was incorporated into the code. The original TWINKLE model was retained in TWINKLE-M, but for licensing analysis the MATPRO model is used. The applicant committed to updating the TWINKLE-M input manual to reflect clearly the source of the MATPRO model in TWINKLE-M and mandate its usage for licensing analysis. The updated MATPRO model is included in Revision 3 of the TWINKLE-M manual (Ref. 11), and is used for licensing analysis. This is acceptable because the MATPRO database is considered an industry standard for fuel specific data, including fuel specific heat.

RAI 3.2-1 asked why the introduction of discontinuity factors was not considered a change to the constitutive models. The applicant presented a figure, Figure 2.2-1.1, "Flowchart of the Discontinuity Factor Process," in its response to RAI 2.2-1 [Ref. 19], showing that the addition of the discontinuity factor can be viewed as a pre-processing and post-processing activity outside the solution algorithm of the TWINKLE-M code and therefore does not represent a change to the constitutive model. The applicant's response is acceptable because it does not change the fundamental diffusion equations used by TWINKLE-M to solve for the neutron flux.

Topical Report Section 3.2 provides the TWINKLE-M code validation of the three-dimensional capability by comparing core power distribution and other parameters with the ANC code [Ref. 17]. A sensitivity study of the radial mesh size is also provided Section 3.2.2. The applicant concluded that a 2x2 mesh per assembly was sufficient for use in accident analysis.

Section 3.2.1 presented a comparison between TWINKLE-M and ANC for cases with and without an ejected rod under steady-state conditions. The three comparison cases presented

are: (1) BOC HFP with all rods out, (2) EOC HZP with all rods at the zero power insertion limit, and (3) EOC HZP with all but one rod inserted. The first two cases are characterized by radial and axial power distributions in the normal operating condition. The third case represents a highly peaked radial power distribution characteristic of an RCCA ejection accident from HZP condition. All comparisons were done at steady state conditions because, although TWINKLE-M is a transient code, ANC is a steady state code. Topical report Figures 3.2.1-2, "Radial Power Distribution Comparison with ANC and TWINKLE-M Case 1, BOC HFP All RCCAs Out," 3.2.1-3, "Radial Power Distribution Comparison with ANC and TWINKLE-M Case 2, EOC HZP RCCA at Insertion Limit," and 3.2.1-4, "Radial Power Distribution Comparison with ANC and TWINKLE-M Case 3, EOC HZP One RCCA Ejected," respectively, show the radial power distribution comparison between the TWINKLE-M and ANC for the three cases analyzed. Topical report Figure 3.2.1.5, "Average Axial Power Distribution Comparison with ANC and TWINKLE-M," shows the TWINKLE-M/ANC comparison of the average axial power distributions for the three cases. The results show good agreement between the two codes.

In reviewing the analysis of the three cases for the TWINKLE-M validation, the staff also requested the applicant to address RAI 3.2-3 through RAI 3.2-6.

RAI 3.2-3 requested an explanation of the treatment of the core-reflector boundary condition. In its response [Ref. 19], the applicant stated that the diffusion coefficient of the reflector region was modified externally with region-specific multipliers so that the TWINKLE-M power distribution agreed with the ANC calculation, both before and after the rod ejection. RAI 3.2-3-1 requested that the applicant confirm that the modification was made to both the radial reflector region and the two axial reflector regions. In its response [Ref. 22], the applicant stated that the fast group diffusion coefficient was modified for all reflector regions. RAI 3.2-3-2 requested a detailed explanation of the modification process. In its response [Ref. 22], the applicant stated
[(Proprietary information withheld under 10 CFR 2.390)
]

RAIs 3.2-4 and 3.2-5 requested an elaboration of the differences between TWINKLE-M and ANC for the HZP all rods in (ARI) and the HFP all rods out (ARO) cases, respectively. In its response [Ref. 19], the applicant stated that the differences were predominantly due to different numerical solution algorithms used in the two codes. The applicant noted that the largest difference between the results with the two codes for the HZP ARI case occurred near the inserted rods, whereas, for the HFP ARO comparison, the largest differences occurred where fresh fuel was adjacent to high burnup fuel. These locations were characterized by steep flux gradients. The nodal expansion method used in ANC calculates steep flux gradients more accurately than the finite difference method used in TWINKLE-M. RAI 3.2-4-1 requested further information regarding burnup differences being responsible for code differences. In its response [Ref. 22], the applicant provided a radial map of assembly burnups for the HFP ARO case and demonstrated that the largest differences between TWINKLE-M and ANC were at locations where adjacent assemblies had much different burnups.

RAI 3.2-6 requested a comparison of the Doppler reactivity coefficient calculated by ANC and by TWINKLE-M for the rod ejection accident. For large ejected rod worths Doppler feedback inserts negative reactivity which limits the reactor power excursion. Therefore, the staff asked how ANC, which is a modern nodal code compares with TWINKLE-M. In its response [Ref. 19], the applicant showed the two calculated values to be identical.

RAI 3.2-2 requested the applicant provide additional verification of TWINKLE-M when used to predict rod ejection behavior. In its response [Ref. 21], the applicant provided a comparison of

radial power distributions of a typical three-loop operating plant to the TWINKLE-M calculations and stated that the good agreement between the two demonstrated the accuracy of the TWINKLE-M 3-D steady state calculation. As verification of the transient capability of TWINKLE-M, the applicant presented the results of a TWINKLE 3-D calculation of the OECD rod ejection benchmark problem. The agreement between TWINKLE-M and the reference solutions was good. Therefore, the staff finds this additional verification acceptable.

Based on the above evaluation, the staff concludes that the TWINKLE-M three-dimensional capability validation is successfully demonstrated by the TWINKLE-M/ANC comparison.

Topical report Section 3.2.2 presents the sensitivity study of the mesh size in the TWINKLE-M analysis of a rod ejection accident in order to determine the optimal spatial mesh size. The sensitivity study is performed for two cases: with an 2x2 meshes and an 4x4 meshes per fuel assembly, respectively, in the radial direction. [(Proprietary information withheld under 10 CFR 2.390)] Topical report Figures 3.2.2-1 and 3.2.2-2 show the comparisons of the nuclear power and hot channel factor, respectively, of the 2x2 and 4x4 mesh results. The results show good agreement between the two mesh configurations of the maximum nuclear power and maximum hot channel factor.

RAI 3.2-7 requested additional information regarding calculated ejected rod worth, fuel temperature, hot channel factors, and axial mesh sensitivity. RAI 3.2-10 also requested the sensitivity of the computed results to axial mesh size. In its response [Ref. 21], the applicant provided the requested information and demonstrated that the 2x2 mesh simulation was in good agreement with the 4x4 simulation. The applicant's response also showed that doubling the axial mesh size resulted in axial power profiles that were generally closer to the ANC calculation but not significantly so. RAI 3.2-7-1 questioned whether the 4x4 results were converged. In its response [Ref. 22], the applicant stated that one would expect a better agreement between the ANC and TWINKLE-M power distributions as mesh size was decreased in TWINKLE. The applicant noted, however, that the key output parameters for the rod ejection accident were not highly dependent on small differences in power distributions. Hence, using a smaller mesh size would not significantly alter the final result.

The time step sizes used in TWINKLE-M are provided in the response to RAI 3.2-7-2 [Ref. 22]. RAI 3.2-7-3 requested why the diffusion coefficient was not adjusted when the axial mesh size was changed. In its response [Ref. 22], the applicant stated that its goal was to show the effect of axial nodalization without any diffusion coefficient adjustment. Had such an adjustment been made, the power traces from the two axial nodalization cases would have been much closer. The responses are acceptable because they demonstrate that the TWINKLE-M model's mesh size is adequate and the results are converged.

RAI 3.2-8 and 3.2-9, respectively, requested why the neutron lifetime and the delayed neutron fraction were listed as a calculation condition in the mesh sensitivity study. In its response [Ref. 19], the applicant explained that inputs to both sensitivity cases were adjusted so that both cases had identical values of neutron lifetime and delayed neutron fraction. When the appropriate adjustments were made, these parameters did not affect the results of the sensitivity study of mesh size for the rod ejection simulations. RAI 3.2-8-1 and RAI 3.2-9-1 asked the applicant to clarify whether the adjustments to neutron lifetime and delayed neutron fraction were done only for the mesh sensitivity study. In its response [Ref. 22], the applicant stated that the adjustments were made as part of each TWINKLE-M analysis so that the parameters' values match the safety limit values. These responses demonstrate that the

parameters were adjusted to the appropriate safety limit values for each transient; and therefore, are acceptable.

The staff recognized that, while the applicant had provided numerous comparisons between TWINKLE-M and other reactor physics computer codes in the topical report and RAI responses, no comparisons had been presented between TWINKLE-M results and experimental data. Therefore, RAI REA-1 requested that the applicant perform such a comparison, or justify why lack of such a comparison is acceptable.

In its response to RAI REA-1 [Ref. 27], the applicant provided a comparison of SPERT III E-Core Test No. 60 (hot startup conditions), as documented in IDO-17281 [Ref. 32]. SPERT III was an experimental reactor kinetics facility operated at the Idaho National Engineering Laboratory in the late 1960's to provide validation data for reactor analysis codes such as TWINKLE. The E-Core configuration consisted of a small, uranium-oxide fueled core arranged in a lattice geometry generally similar to operating pressurized water reactors, with representative coolant flow rates and pressures. [

(Proprietary information withheld under 10 CFR 2.390)

]

The applicant presented the core geometry modeled in the TWINKLE-M code to represent the SPERT III reactor, and the experimental conditions at the time of the test. [

(Proprietary information withheld under 10 CFR 2.390)

]

In a follow-up for RAI REA-1, the staff requested in RAI REA-9, that the applicant provide the source of the nuclear data (cross sections, diffusion coefficients, etc.) used to perform the simulation of SPERT-III E-Core Test No. 60.

In its response to RAI REA-9 [Ref. 33], the applicant provided electronic files containing the requested nuclear data. [

(Proprietary information withheld under 10 CFR 2.390)

]

The applicant provided the requested validation of the TWINKLE-M code using data from the SPERT III experiment, which demonstrates satisfactory agreement between calculated and measured data. The applicant also provided the requested source of the nuclear data used in the TWINKLE-M simulation of the experiment. Therefore, the staff finds the responses to RAI REA-1 and RAI REA-9 acceptable.

In summary, the staff evaluated the TWINKLE-M code description provided in Section 2.2 and the TWINKLE-M validation presented in Section 3.2 of Topical Report MUAP-07010-P (R2) [Ref. 3]. The staff identified the key issues related to this area of review and issued numerous RAIs. The applicant responses satisfactorily resolved all of the staff's concerns addressed in the RAIs. Based on comparisons between TWINKLE-M and the applicant's core design code ANC, comparisons between TWINKLE and other reactor kinetics codes, and the comparisons of TWINKLE-M calculations with the SPERT III experiment measured data, the staff concludes that the verification and validation matrix of the TWINKLE-M code is sufficient to demonstrate that TWINKLE-M is capable of adequately simulating the transient response of a reactor core to a large reactivity insertion.

5.1.4 VIPRE-01M

VIPRE-01M [Ref. 12] is the applicant's version of VIPRE-01 [Ref. 15], which is a subchannel analysis code used for thermal-hydraulic analyses in reactor cores. VIPRE-01M is primarily used to evaluate the reactor core thermal limit, the minimum departure from nucleate boiling ratio (MDNBR). In the Non-LOCA methodology, VIPRE-01M is used together with TWINKLE-M for the analysis of the uncontrolled control rod assembly withdrawal from subcritical and control rod ejection events. VIPRE-01M is also used in Non-LOCA methodology for the analysis of other events whenever the computed MDNBRs from MARVEL-M are outside the range of the MDNBR tables built into MARVEL-M. The acceptability of VIPRE-01M for use in the Non-LOCA methodology is addressed in a separate staff review of the applicant's thermal design methodology in [Ref. 12].

Condition. The Non-LOCA methodology described in MUAP-07010 Revision 4 is acceptable provided that it includes an approved method to model detailed core sub-channel thermal-hydraulic conditions. For the US-APWR DC application, that method is based on VIPRE-01M as documented in MUAP-07009 which is under NRC review.

5.1.5 ANC

PARAGON/ANC [Ref. 17] is an NRC approved core design code system.

As discussed in the topical report [Ref. 53] and in Sections 4.0, 5.1.3, and 5.2 of this SER, the ANC code and results obtained with it are used by the applicant as a basis for models, the validation and assessment of models, and the checking and generating of core reactivities and power distributions. As discussed in Section 5.2 of this SER, ANC is used to provide a standard for establishing the TWINKLE-M core initial power distributions for the Spectrum of RCCA Ejection event, and in combination with MARVEL-M in the analysis of the Inadvertent Opening of an SG Relief or Safety Valve and Steam Line Piping Failure events, for which it is used to calculate local peaking factors if the reactor is determined to return to power following scram.

The staff's evaluation finds that ANC is acceptable for use in the US-APWR Non-LOCA methodology for DCD Tier 2 Chapter 15 analyses based on the staff's review documented in the DCD Chapter 4 SE. [Ref. 4].

5.2 Event-Specific Methodology

This SER section evaluates the US-APWR event-specific Non-LOCA methodology, which is presented by the applicant in Section 5.0 of the topical report [Ref. 53].

The staff grouped the Non-LOCA events into four categories, based on the computer codes used in the event analyses. The four categories are given below, along with the events the applicant selected for presenting detailed descriptions of the application of its Non-LOCA methodology, including the use of sample calculations:

- Analyzed using MARVEL-M only
  - Uncontrolled RCCA Bank Withdrawal at Power

- Analyzed using MARVEL-M and VIPRE-01M in sequence
  - Complete Loss of Forced Reactor Coolant Flow

- Analyzed using TWINKLE-M and VIPRE-01M in sequence
  - Spectrum of RCCA Ejection

- Analyzed using special code models in MARVEL-M
  - Steam System Piping Failure
  - Feedwater System Pipe Break
  - Steam Generator Tube Rupture

It is noted that the sample transients selected by the applicant and presented in Section 6 of the topical report are for the purpose of illustrating the detailed event-specific analysis methods described in Section 5 of the topical report. These illustrations facilitate the evaluation of the Non-LOCA methodology, which is the subject of this SER. The staff evaluation of the safety analyses for all accident and transient events is documented in a separate SER that reviews Chapter 15 of the US-APWR DCD [Ref. 4].

The staff's review of topical report Section 5.0 began with consideration of the common elements of the methods employed by the applicant for the analysis of the Non-LOCA transient and accident events.

In its response to RAI Gen-2 [Ref. 36], the applicant noted that the core pressure was restricted to an upper limit on plotted input for those analyses which use the DNBR lookup tables in MARVEL-M. The upper limit was set by user input variable [

(Proprietary information withheld under 10 CFR 2.390)

]

The value given in DCD Tier 2 Table 4.1-1 is the value which conservatively bounds the uncertainties related to the bypass flow estimation. The best estimate value is determined based on a statistical combination of uncertainties related to bypass flow as described in [Ref. 37]. The applicant's response explains how the bypass input value for MARVEL-M was derived and why it differs from the DCD Tier 2 Table 4.1-1 value. Since the value being used is appropriate, the response to RAI Gen-8 is acceptable.

To calculate the maximum RCS pressure for overpressure events, the applicant adds the absolute value of [

(Proprietary information withheld under 10 CFR 2.390)

] The response is acceptable because it demonstrates that MARVEL-M adds a conservative bias to its maximum calculated pressure in the RCS.

The staff's evaluations of the application of the Non-LOCA methodology to each of the events described in Section 5 of the topical report are presented below.

5.2.1 Uncontrolled RCCA Bank Withdrawal at Power

The applicant presented the specific analysis methodology and a sample analysis for this event in Sections 5.1 and 6.1, respectively, of the topical report.

The uncontrolled RCCA bank withdrawal at power is categorized as an AOO. It results in an increase in power and reactor coolant temperature. Unmitigated, the increases in core power and coolant temperature would eventually reach a DNB, overpower, or RCS pressure limit. This AOO is mitigated, however, by the following Reactor Protection System (RPS) trips, only two of which were credited in the analysis:

- Neutron flux high trip (credited)
- Neutron flux rate high trip
- Over power delta-T high trip
- Over temperature delta-T high trip (credited)
- Pressurizer pressure high trip
- Pressurizer water level high trip

This event is analyzed using only MARVEL-M. Minimum DNBR is calculated by the MARVEL-M code using DNBR data tables input to MARVEL-M. (The staff review of the MARVEL-M DNBR model is addressed in Section A.2.4 of Appendix A of this report.) The data tables were constructed by performing several VIPRE-01M calculations with constant core flow and various core heat fluxes, core inlet temperatures, and pressures. A comparison of DNBR calculated by the tables in MARVEL-M to DNBR calculated by VIPRE-01M was presented in MUAP-07010, Appendix A of [Ref. 1]. In its response to RAI App.A-1 [Ref. 19], the applicant clarified that the DNB tables in MARVEL-M were generated assuming a constant core inlet flow rate and a constant core power distribution. Therefore the DNB tables can be used for each AOO which has a constant core flow and a power distribution bounded by the distribution used

in generation of the table. The applicant further noted that the comparison in Appendix A demonstrates that the MARVEL-M interpolation methodology adequately calculated DNBR for the uncontrolled RCCA bank withdrawal at power event. The implementation of the DNBR tables into MARVEL-M is acceptable to the staff.

The Non-LOCA methodology for this analysis requires the use of conservative trip delays and a conservative reactor scram insertion curve. The event is evaluated at both BOC and EOC conditions, and over a range of reactivity insertion rates up to the maximum insertion rate of 75 pcm/s (percent mili/s). The staff reviewed the sample calculation and found it to be reasonable because the behavior of key parameters was similar to that seen in the analysis of this event in other reactors using different computer codes.

In summary, the assumptions and application of the Non-LOCA methodology and the results of the sample calculation of the RCCA Bank Withdrawal at Power event were reviewed. The calculated results were reasonable. Therefore, the staff finds the applicant's Non-LOCA methodology acceptable for evaluating the US-APWR RCCA Bank Withdrawal at Power.

5.2.2 Complete Loss of Forced Reactor Coolant Flow

The applicant presented the specific analysis methodology and a sample analysis for this event in Sections 5.2 and 6.2, respectively, of the topical report [Ref. 53].

The complete loss of flow (CLOF) event is an AOO which results from a loss of electrical supply to the reactor coolant pumps. If the reactor is at power when the event occurs the loss of flow results in a rapid increase in core temperature and an erosion of the DNBR margin. The Non-LOCA methodology assumes the following RPS trips are available to provide protection:

- Reactor coolant pump low speed trip
- Reactor coolant flow low trip

This event is analyzed using MARVEL-M and VIPRE-01M. The reactor system response is calculated using MARVEL-M. The DNBR tables in MARVEL-M are not used for this event because they are based on constant core flow; core flow is decreasing in this event. MARVEL-M generates a file of time-dependent core flow and power for input to VIPRE-01M. The VIPRE-01M code calculates the MDNBR assuming a constant design limit enthalpy rise in the hot channel and a design limit axial power distribution. For conservatism the increase in core pressure during the event is not credited in the VIPRE-01M calculation. Conservative values of RCP inertia, trip delays, scram insertion, and reactivity coefficients are used.

In its response to RAI 5.2-1 [Ref. 19], the applicant performed a sensitivity study to show the effect of RCP inertia. The CLOF calculation was done using the design value of RCP inertia and compared to the base calculation which used a conservative value of RCP inertia. The study demonstrated that the rate of flow coast-down was faster when the conservative RCP inertia was used and the MDNBR was slightly lower. The response demonstrates that the value used for the RCP inertia was conservative; and is therefore, acceptable.

RAI 5.2-2 requested details regarding the DNB correlation used for the CLOF analysis. In its response [Ref. 19], the applicant stated that the US-APWR analysis applies the WRB-2 correlation and referred to Topical Report MUAP-07009, "Thermal Design Methodology"

[Ref. 12] for detailed information on the DNB correlation. The requested information was located in the topical report; therefore, the response to RAI 5.2-2 is acceptable.

In its response to RAI 6.2-1 [Ref. 19], the applicant explained that the revised thermal design procedure (RTDP) is used to statistically account for uncertainties in the input parameters such as reactor power, RCS pressure, flow rate, and average temperature; and conservative values are used for other parameters such as reactor trip simulation and reactivity coefficients. Since the RTDP has been approved by NRC [Ref. 55], the staff finds the uncertainty treatment to be acceptable.

The staff performed an independent mass balance on the pressurizer for the CLOF sample problem and found that mass conservation was violated. In its response to RAI CLOF-2 [Ref. 36], the applicant noted that the parameters used in the mass balance calculation were misidentified in the input manual and were not what they were believed to be in the staff's independent calculation. The description was corrected in the MARVEL-M theory manual. The applicant recreated the mass balance calculation using pertinent parameters and showed that mass conservation was not violated. The response is acceptable because the MARVEL-M theory manual was corrected to show mass conservation.

When the steam and liquid temperatures for the CLOF sample problem were reviewed it was noted that the initial steam temperature indicated subcooled steam. RAI CLOF-3 requested the applicant to explain why the steam temperature was below saturation. In its response [Ref. 36], the applicant stated that the request was about parameters that are only used for information; they are not used in internal calculations. The response is acceptable because the temperature discrepancy noted by the staff has no effect on the MARVEL-M calculations.

Since the CLOF sample problem had no fluid leaving or entering the primary system it was used to check mass and energy conservation in MARVEL-M. Using Excel spreadsheets and the information from MARVEL-M edits, the staff calculated the primary system mass and energy at 0, 6, and 10 seconds. The evaluation indicated that approximately 5 percent of the total energy had disappeared by 6 seconds and 10 percent by 10 seconds. RAI CLOF-4 requested the applicant to explain the apparent non-conservation of energy in the MARVEL-M calculation. In its response [Ref. 36], the applicant noted that the calculations had omitted a term used to account for the energy stored in the SG tube metal. The applicant explained how MARVEL-M accounts for this stored energy and modified the staff spreadsheets to include the metal stored energy. When this was done, the difference in the primary system total energy at 0 and 6 seconds was 1 percent, and the difference between the energy at 0 and 10 seconds was 0.7 percent.

These differences were within the uncertainty of the calculations. The staff reviewed the calculations as modified by the applicant, and confirmed the applicant's conclusion that MARVEL-M provides a reasonable conservation of mass and energy.

In its response to CLOF-5 [Ref. 36], the applicant noted that the SG tube metal mass calculation is controlled by the input variable [

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] and is acceptable.

In summary, the staff reviewed the application of the Non-LOCA methodology and the results of the sample calculation of the CLOF event which are reasonable and conservative. Therefore, the staff finds the Non-LOCA methodology acceptable for evaluating the US-APWR CLOF event.

5.2.3 Spectrum of RCCA Ejection

The applicant presented the specific analysis methodology and a sample analysis for this event in Sections 5.3 and 6.3, respectively, of the topical report [Ref. 53].

The RCCA ejection is a PA. An assumed failure of a control rod drive mechanism pressure housing results in the rapid ejection of the RCCA and its drive shaft. The RCCA ejection leads to a rapid positive reactivity insertion and an increase in local power peaking and possible localized fuel failure. The RCCA ejection event nuclear excursion is terminated by Doppler reactivity feedback from increased fuel temperature. The following RPS trips are available to trip the reactor for RCCA ejection events; trips that are not credited are identified:

- Neutron flux high trip - high setting
- Neutron flux high trip - low setting
- Neutron flux rate high trip (not credited)
- Over temperature ΔT trip (credited only for events where power does not reach setpoint for neutron flux high trip)
- Low pressurizer pressure trip (credited only for events where power does not reach setpoint for neutron flux high trip)

The analysis method described in Section 5.3 of the topical report uses primarily TWINKLE-M and VIPRE-01M to analyze the RCCA ejection event. TWINKLE-M is used to determine the transient core average and local power behavior. For HZP conditions a three-dimensional model of the core is used in TWINKLE-M. For HFP conditions a one-dimensional TWINKLE-M model is used. VIPRE-01M calculates the fuel centerline temperature, cladding temperature, fuel enthalpy, and DNBR. For the HZP analysis the VIPRE-01M calculation uses two interface files created by TWINKLE-M; one is the time-dependent core average power and the second is the hot channel power factor. For the HFP event the VIPRE-01M calculation uses only the time-dependent average core power interface file. The hot channel peaking factor is assumed constant at the design limit for the HFP VIPRE-01M calculation. The VIPRE-01M analysis uses a one-eighth core model in which each subchannel of the hot assembly is represented.

For large reactivity insertions which could potentially challenge fuel centerline melt or PCMI criteria the HFP, 1-D TWINKLE-M cases insert the design limit reactivity within 0.1 seconds.

The reactivity insertion is accomplished by modifying the TWINKLE-M eigenvalue to yield the design limit RCCA reactivity worth. As described in the staff's evaluation of RAIs 5.3-1-3 [Ref. 22] and RAI 5.3-1 [Ref. 19] below, this was found to be an acceptable way of inserting the design limit reactivity. For the HZP, 3-D analysis the ejected rod worth is increased by changing the local absorption cross section. If the design limit reactivity is not met the eigenvalue is modified to reach the design limit reactivity. Adjusting the eigenvalue, if the design limit reactivity is not met, is conservative as the localized Doppler feedbacks would be larger if the RCCA worth was increased as described in the response to RAI 5.3-3 below [Ref. 19].

For both HFP and HZP cases Doppler reactivity feedback is reduced by [

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] is conservative for determining the maximum fuel centerline temperature, fuel enthalpy and fuel enthalpy rise.

In RAI REA-8 and REA-12.2, the staff asked the applicant if low ejected rod worths which don't reach the high flux trip point had been evaluated. In its response to RAI REA-12 [Ref. 33], the applicant described a methodology which addressed rod ejections which don't reach the high flux setpoint and depressurize the RCS (note that the methodology in the response to REA-8 is not used). Low rod worth ejections could yield a higher number of DNB fuel failures as reactivity feedbacks are not as great and the reactor trip either does not occur or is delayed relative to high flux trip. The applicant analyzed low ejected rod worths by using the methodology described in the RAI REA-12.2 response by evaluating the accident with three, HFP, bounding steady state scenarios. The first scenario evaluated the short-term effects at peak core power and peaking factor conditions while holding thermal-hydraulic conditions constant. The second scenario evaluated a long-term, rapid RCS depressurization, while the third evaluated a long-term, slow RCS depressurization. The rapid and slow RCS depressurization cases correspond to different RCS hole size assumptions. The 3-D TWINKLE-M, VIPRE-01M and ANC codes were used in the evaluations.

The short term analysis assumes that RCS pressure remains constant; [

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] The total numbers of failed rods are determined and compared to the assumed value in DCD Chapter 15.4.8.

Several RAIs were issued regarding the use of a 1-D vs. 3-D TWINKLE-M model and the use of a 1/8 core VIPRE-01M model for the rod ejection analyses.

In its response to RAI 5.3-6 [Ref. 19], the applicant stated that the 1/8 core model is used simply to assure consistency of the VIPRE-01M model input with the core thermal design model. In its response to RAI 5.3-6 [Ref. 19], and RAI 5.3-6-1 [Ref. 22], the applicant explained that the 1/8-core representation used in the VIPRE-01M model allows flow redistribution from the hot assembly. This flow redistribution results in a more limiting coolant condition in the hot assembly. The staff finds the VIPRE-01M model acceptable for the intended analyses.

RAI 5.3-9 requested the applicant explain why the 3-D TWINKLE-M model wasn't used for both the HZP and the HFP rod ejection analyses. In its response [Ref. 19], the applicant stated that the 3-D model was necessary to properly capture the effect of Doppler feedback in an event where the power distribution was highly skewed. Since the rods were mostly out of the core for the HFP event the power distribution was much less skewed in the HFP event relative to the HZP event. Therefore, there was no real advantage to using the 3-D model for the HFP event analysis. The applicant chose to use the 1-D model to simplify the analysis. RAI 5.3-7 requested an explanation of how the TWINKLE-M 1-D model was obtained and how the 1-D model's axial power distribution compared to the 3-D models. In its response to this RAI [Ref. 19] and followup RAI 5.3-7-1 [Ref. 22], the applicant showed how the macroscopic cross sections, diffusion coefficients, and delayed neutron data were collapsed from 3-D to 1-D. The applicant also presented a comparison of the two models' axial power distribution for BOC, HFP, ARO conditions and concluded that the two models agreed very well. The response to RAI 5.3-7-2 [Ref. 22] noted that small differences in axial power shape would have no significant impact on the number of rods calculated to be in DNB. Based on the axial similarities of the 3-D and 1-D axial power shapes the staff agrees that a 1-D TWINKLE model is acceptable for the HFP analysis where rod insertion limits prevent a large radial power distribution distortion. A 3-D analysis at HZP is acceptable as the radial power distribution is more skewed due to the deeper rod insertion. One of the analysis assumptions made for the reactivity feedback of the RCCA ejection calculation is to apply a conservative multiplier on the fast absorption cross section to yield a conservative Doppler feedback. RAI 5.3-8 requested the applicant to elaborate on the conservative multiplier. In its response [Ref. 19], the applicant provided the equation for adjusting the fast absorption cross section and showed a conservative factor was applied to the change in the calculated effective fuel temperature yielding a conservative Doppler feedback. The staff agreed that reducing the fast absorption cross section using the applicant's method will reduce the Doppler feedback making the transient analysis results more conservative.

The important parameters for the RCCA ejection analysis are fuel temperature, PCMI, energy deposition to the fuel, and DNBR. The applicant's methodology applies conservatisms for the RCCA ejection analysis depending on the figure of merit. First, a conservatively large reactivity, the design limit, is inserted within 0.1 seconds. In HZP cases the inserted reactivity is calculated directly by changing the local absorption cross section caused by the ejection of the most reactive RCCA. In its response to RAI 5.3-1 [Ref. 19], the applicant explained that because this directly calculated reactivity was less than the design reactivity, the difference between it and the design value was inserted by linearly changing the eigenvalue in the 3D TWINKLE-M calculation. This is an appropriate conservative procedure and is acceptable. Additional conservatism is applied in the treatment of the hot channel factors. In the HZP analysis the hot channel power factors from TWINKLE-M are scaled by a multiplier so that the maximum value passed to VIPRE-01M is at the design limit. VIPRE-01M then calculates a maximum fuel centerline temperature, the maximum fuel enthalpy and an adiabatic fuel enthalpy rise for comparison to the appropriate safety limit values. RAI 5.3-2 and RAI 5.3-2-3 requested the applicant explain why the important safety parameters were not obtained directly from TWINKLE-M instead of VIPRE-01M. In its responses to RAI 5.3-2 [Ref. 19] and RAI 5.3-2-3 [Ref. 22], respectively, the applicant explained that the TWINKLE-M calculation uses a [(Proprietary information withheld under 10 CFR 2.390)] It passes its mesh-wise fuel enthalpy to VIPRE-01M. VIPRE-01M then constructs the hot assembly, pin-by-pin enthalpy distribution. The staff finds this approach acceptable as it conservatively sets all 17 hot assembly pin powers to the design limit value.

In its response to RAI 5.3-3 [Ref. 19], the applicant noted that the hot channel factor from TWINKLE-M run was not at the design limit power because the margin associated with the design limit value is not accounted for in TWINKLE-M. This is conservative for the core kinetics calculation because the lower hot channel factor decreases the local Doppler feedback. By raising the pin power to the design limit in VIPRE-01M a conservative calculation of hot spot fuel temperature and enthalpy is obtained. The response to RAI 5.3-8 [Ref. 19] noted another conservative feature in the Doppler feedback calculated by TWINKLE-M: A 20 percent reduction in the calculated increase of the fast group absorption cross section caused by a fuel temperature change. The responses are acceptable because they explain the incorporation of conservatisms into the RCCA ejection accident analysis.

PCMI fuel failure is evaluated by first calculating the local adiabatic fuel enthalpy rise, ΔH, at each mesh point in TWINKLE-M. The peak TWINKLE-M mesh power is then adjusted upward to the maximum hot spot enthalpy in VIPRE-01M. This adjustment (ratio) is then applied to determine all the TWINKLE-M mesh enthalpies. Since the burnup distribution is specified in TWINKLE-M, a relationship between the burnup and ΔH can be established. The relationship between local oxide thickness to cladding wall thickness and the local burnup is available from fuel design calculations. This is used, along with the burnup/ΔH relationship, to establish the ΔH versus oxide/wall thickness distribution for the core. Finally, this distribution is compared to the PCMI fuel failure criteria given in SRP Chapter 4.2 Revision 3 Appendix B, Figure B-1, "PWR PCMI Fuel Cladding Failure Criteria." If any rod in a TWINKLE-M mesh point exceeds the failure criteria then all rods within that mesh point are assumed to fail.

RAI 5.3-1-3 requested the applicant show the impact of changing the eigenvalue of the 3D TWINKLE-M calculation rather than changing local absorption cross sections to add reactivity to the 3D TWINKLE-M calculation as described in Section 5.3 of the Non-LOCA topical report. In its response [Ref. 22], the applicant presented the results of two HZP RCCA ejection cases. In [

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] The response to RAI 5.3-2-2 confirms that VIPRE-01M uses the design limit hot channel factor which is conservative when determining fuel centerline temperature and fuel average enthalpy.

In its response to RAI 6.3-1 [Ref. 19], the applicant clarifies the role of the ANC computer code [Ref. 17] in the rod ejection accident. It states that ANC is used to calculate power peaking factor for the rod ejection accident and to calculate the core radial power distribution for the

steam line break (SLB) event. Since ANC is an NRC-approved code, its usage in calculating power peaking factor is acceptable.

RCS pressure associated with the RCCA ejection is calculated using MARVEL-M and VIPRE-01M. VIPRE-01M generates an interface file containing the time-dependent total core void fraction and core heat flux. MARVEL-M reads this file and calculates the resulting RCS pressure transient.

Prior to performing a transient rod ejection analysis, it is necessary to initialize the simulated core at steady-state conditions with respect to core power distribution. For this initialization, the applicant uses a method in which the TWINKLE-M-calculated steady-state power distribution is compared with the power distribution calculated using steady-state core simulator ANC [Ref. 17]. ANC uses a higher-order nodal method to solve the neutron diffusion equation relative to the coarse-mesh finite difference technique used by TWINKLE-M. Therefore, the ANC-generated power distribution is expected to be closer to reality than the raw TWINKLE-M calculation, especially in regions where the flux gradient is the strongest (for example, near the core periphery/heavy neutron reflector interface). In order to improve the agreement between the TWINKLE-M and ANC power distributions, the applicant manually and iteratively adjusts the fast-neutron group nuclear data in the reflector regions of the TWINKLE-M core model. The staff was concerned about the effect of performing such an adjustment at steady-state conditions and expecting it to remain valid for transient conditions, since the flux-weighted cross sections of the peripheral assemblies and the reflector region will change as the power distribution becomes perturbed by a localized reactivity insertion. To address this concern, RAI REA-4 requested that the applicant justify the TWINKLE-M initialization approach used for the analysis of transient events.

In its response to RAI REA-4 [Ref. 27], the applicant provided additional details regarding the adjustment methods used. In short, separate modifications are made to the reflector regions on the top, bottom, flat, and corner regions of the reflector in the form of multipliers to the fast neutron group diffusion coefficient. The applicant stated that once these adjustments are made for a particular core configuration at steady state conditions, they remain valid regardless of transient changes in the core power distribution.

To illustrate this point, the applicant provided reference to a comparison of TWINKLE-M and ANC calculated static power distributions at EOC HZP conditions presented in Topical Report MUAP-07010-P. For this case, the reflector adjustments were made in the TWINKLE-M model with the control rods at their insertion limits to match the ANC model power distribution within approximately [

(Proprietary information withheld under 10 CFR 2.390)

] percent. The applicant concluded that this demonstrates that the adjustments made at the initial conditions remain valid for conditions in which the core flux profile is strongly skewed.

Additionally, the applicant provided a number of mesh sensitivity studies demonstrating that if the axial and radial meshes in the TWINKLE-M model are refined, reflector data adjustments are no longer necessary to bring the TWINKLE-M and ANC power distribution into acceptable agreement. A series of TWINKLE-M HZP transient simulations was provided in which the results using the base simulation's "coarse mesh" model (2x2 radial nodes per assembly,

[
(Proprietary information withheld under 10 CFR 2.390)

] where no reflector adjustments were necessary. The results from the three cases show excellent agreement in ejected rod worth, maximum core power, and maximum hot spot relative power. The applicant concluded that this comparison demonstrates that the reflector adjustments made at steady-state conditions are not invalidated as the core transient progresses.

As a final demonstration of the insensitivity of the reflector adjustments on the core power distribution, the applicant provided a comparison between the base case HZP rod ejection transient, using licensing methodology assumptions as documented in the topical report, with a total of four sensitivity cases. In the sensitivity cases, the initial power distributions were skewed relative to the base case towards the edge of the core, the interior of the core, the top of the core, and bottom of the core. The power distributions were skewed by adjusting the appropriate reflector multiplier relative to the base case. The same ejected rod worth was used in all calculations. The applicant provided the resulting maximum core power and maximum hot spot relative power calculated by TWINKLE-M, the maximum hot spot relative power used in the subsequent VIPRE-01M analysis, and the resulting fuel enthalpy rise. The applicant concluded that, even though some of the sensitivity cases demonstrated a greater core power transient and maximum relative local power as calculated by TWINKLE-M, the cases with higher power had a lower local peaking factor, and vice versa. Additionally, the applicant stated that the conservatism in the licensing methodology (increasing the hot channel power factor to the design limit) when transferring TWINKLE-M results to the VIPRE-01M model is sufficient to bound any effects due to minor variations in the initial core power distribution.

The staff finds the applicant's adjustment of the radial reflector neutron data in order to better the agreement between the TWINKLE-M and ANC steady-state power distributions acceptable. This position is based on the analyses presented by the applicant which show very little impact of the reflector modifications on the calculation of a highly-skewed radial power distribution; sensitivity calculations which show convergence of the reflector modifications to unity as the core spatial mesh is refined; and comparisons between the licensing methodology and sensitivity cases demonstrating abundant margin in the calculated figure of merit (in this case, fuel enthalpy rise) even under conservative analysis conditions

In order to demonstrate compliance with the allowable fuel enthalpy rise related to PCMI in SRP, Section 4.2, Appendix B, the applicant's rod ejection accident (REA) analysis must be performed with the distribution of cladding corrosion within the reactor core. In Section 5.3 of the topical report, the applicant provided a high-level overview of the process used that was not sufficiently detailed for the staff to make a conclusion regarding its acceptability. RAI REA-5 requested more detail from the applicant about the method used.

In its response to RAI REA-5 [Ref. 27], the applicant describes the method by which the nodal fuel enthalpy rise is evaluated and correlated to nodal burnup. First, the nodal enthalpy rise is calculated in the TWINKLE-M code by integrating the local power and power density in each node up to a pre-determined point in the transient.

Then, the node-wise enthalpy rise is increased using the rod-wise peak-to-average ratio within the mesh obtained from the VIPRE-01M hot-spot results. The detailed nodal burnup distribution based on the equilibrium cycle is obtained from the reactor core design code ANC. Thus, for

each node in the TWINKLE-M/VIPRE-01M mesh, a corresponding local burnup from ANC is assigned.

Next, the applicant described the process in which a relationship between local burnup and oxide-to-wall thickness is obtained using the fuel design methods. Due to the axial temperature gradient present in a PWR core, the axial dependence of the corrosion thickness must be evaluated. The applicant states that several potential power histories are calculated in order to determine the limiting case. [(Proprietary information withheld under 10 CFR 2.390)]

Using the results described above, the applicant conservatively determines the relationship between nodal enthalpy rise from the REA calculations and local oxide-to-wall thickness from the fuel design calculations. The resulting relationship is plotted against the SRP 4.2 Appendix B acceptance criterion to verify compliance. For the US-APWR, the applicant states that the maximum enthalpy rise resulting from the REA simulation is less than the minimum value permitted for any oxide-to-wall thickness. Therefore, even with conservative analysis methods, the US-APWR will meet the criterion with sufficient margin.

In follow-up RAI REA-11, the staff requested more detailed information on the method used for increasing the enthalpy rise within the mesh calculated with TWINKLE-M using the rod-wise peak-to-average ratio inside the mesh from the VIPRE-01M hot spot calculation results. In its response to RAI REA-11 [Ref. 54], the applicant provided an example of the method employed using the data from base case TWINKLE-M and VIPRE-01M calculations described in Table REA-4.6 [Ref. 27]. The applicant explained that the enthalpies calculated by TWINKLE-M are compensated so that the maximum value matches the VIPRE-01M value. The hot spot enthalpy result calculated by VIPRE-01M is obtained by using the F_Q transient obtained from TWINKLE-M multiplied by the factor used to increase the peak-to-average ratio inside the mesh as the VIPRE-01M input. The applicant also provided ANC-calculated peak-to-average power data and showed how increasing the mesh-wise maximum hot channel factor obtained by TWINKLE-M in this manner conservatively bounds the mesh nodal powers.

Based on the elaborated discussion of the evaluation process described above, the staff finds the method in which the applicant demonstrates compliance with the SRP 4.2 Appendix B criterion regarding enthalpy rise as a function of oxide-to-wall thickness to be acceptable. The staff finds the method employed by the applicant to make this determination to be suitably conservative and to appropriately account for fuel burnup effects.

The staff determined that the initial axial power distributions used for analysis of the REA event had not been discussed in the topical report or prior RAI responses. The staff therefore issued RAI REA-6 requesting the applicant describe the axial power distributions used, the basis for their selection, and the sensitivity of the REA simulation results to variations in the initial axial power distribution.

In its response to RAI REA-6 [Ref. 27], the applicant stated that the best-estimate, equilibrium cycle power distribution, calculated by ANC, is used. As described above in the response to RAI REA-4, adjustments are made to the TWINKLE-M model to bring the TWINKLE-M and ANC power distributions into satisfactory agreement.

The applicant provided a list of conservatisms in the TWINKLE-M calculations that would compensate for any non-conservatism in the initial axial power distribution. These conservatisms include adjusting the reactivity insertion to the maximum design limit, using [(Proprietary information withheld under 10 CFR 2.390)] design value, and minimizing the

delayed neutron fraction and neutron lifetime. The applicant cited additional conservatisms in the core thermal analysis using the VIPRE-01M code, such as adjusting the local peaking factors to the design limit.

For the hot zero power analysis cases, the applicant referred to Appendix B of the topical report for a quantification of the conservatism inherent when using the above assumptions. The applicant also referred to the response to RAI REA-4, which demonstrates little sensitivity of the transient core power to the initial axial power distribution.

For the hot full power analysis cases, the applicant states that, depending on which limiting parameters are of interest in the thermal analysis, different axial power distributions are employed. For the fuel temperature analysis the VIPRE-01M model uses a chopped-cosine axial power distribution, while for the rods-in-DNB analysis the VIPRE-01M model uses a double-hump axial power distribution. Both axial power profiles are provided in DCD Figure 4.4-4. The applicant has shown for the thermal analysis of the reactor core that these power distributions are limiting.

Based on the conservatism of the assumptions used in the TWINKLE-M neutronics analysis (which maximizes the core power transient in both the hot zero and hot full power cases) and the conservatisms inherent in the thermal analysis methodology (which includes adjusting local peaking factors to design limits), the staff finds the treatment of the initial power distribution employed by the applicant for the REA analysis to be acceptable.

For the hot full power REA analysis, the one-dimensional methodology involves simulating the reactivity insertion resulting from control rod ejection by directly modifying the effective multiplication constant in the transient diffusion equation and not by simulating the change in nodal cross sections resulting from the control rod exiting the core. As a result, the power transient is simulated globally (i.e., core-wide) rather than locally. In RAI REA-7, the staff requested that the applicant explain how the local effects of the rod ejection are accounted for in the methodology.

In its response to RAI REA-7 [Ref. 27], the applicant stated that, while from a reactivity-insertion point of view the global axial power distribution does not change, there is some effect on the axial power shape prior to reactor trip due to feedback effects. However, the methodology does directly simulate the effect of reactor trip control rod insertion by changing the nodal cross sections. This causes the power shape to be significantly skewed toward the bottom of the core as the reactor trip reactivity is inserted.

The applicant acknowledged that simulating a reactivity insertion globally does not directly account for the effect of the rod ejection on axial flux shape but also indicated that the effect is minor for the hot full power cases. In its response to RAI-REA 6 [Ref. 27], the applicant stated that any non-conservatisms in the reactivity insertion are well-bounded by other conservatisms in the methodology.

To further support the applicant's position that other methodology conservatisms cover any effect of different axial power distributions, the applicant referenced the response to RAI 15.4.8-5 [Ref. 40]. Within this RAI response, best-estimate three-dimensional REA simulations were performed at HFP conditions and compared with the one-dimensional licensing methodology described in the topical report.

In each case, the resulting core power and fuel temperature transients were much more severe when using the one-dimensional licensing method than when using the three-dimensional, best-estimate method. The applicant indicated that the response to RAI 15.4.8-5 demonstrates the conservatisms in the one-dimensional modeling approach, which more than compensate for the effects of variations in the initial core axial power distribution.

Based on the above discussion and the analysis presented by the applicant, the staff finds the method by which the applicant simulates the REA from hot full power conditions and the applicant response to RAI REA-7 to be acceptable. This finding is based primarily on the conservatisms employed in the core thermal analysis (namely, increasing the local peaking factor to the design limit and maintaining it at the design limit for the duration of the transient), as well as the comparisons presented by the applicant between the licensing methodology and the best-estimate calculations.

The high worth rod ejection analysis methodology adds sufficient positive reactivity to reach the high flux reactor trip setpoint. The nominal trip setpoint is 109 percent and for conservatism the methodology adds an additional 9 percent. The analysis is performed assuming a setpoint of 118 percent. The staff agreed that this approach is conservative from a view of prompt energy deposition but questioned if it is also conservative from a view of rods-in-DNB. The staff issued RAI REA-8 requesting the applicant perform an analysis or evaluation where the positive reactivity addition causes the flux to get very close to, but not reach, the 118 percent trip setpoint. Of interest to the staff for such cases are what other reactor trips may occur, the reactor trip time, the percentage of rods in DNB.

As a result of considering issues related to the high flux reactor trip setpoint discussed above, the staff questioned if the 9 percent increase in the trip setpoint (from 109 percent nominal to 118 percent used in the analysis) is sufficient to bound all accident scenarios and issued RAI REA-12 requesting the applicant address the bases for the 9 percent uncertainty. As an example, the staff suggested a case where of the four ex-core flux detectors, one is out of service, another neighboring one is assumed to fail, and the ejected rod is on the core periphery mid-way between the two inoperative detectors.

In its response to RAI REA-12.1 [Ref. 42], the applicant stated that the 9 percent increase may not bound the uncertainty for all possible cases, especially considering the locations of ex-core detectors relative to the ejected rod. The power increase seen by the operable detectors may not reach the 118 percent analytical setpoint. The applicant stated that in that event, reactor trip will result from the over temperature delta temperature or low pressurizer pressure conditions, assuming RCS depressurization following break of the control rod drive mechanism (CRDM) housing. The applicant evaluated the uncertainty associated with the ex-core detectors and performed sensitivity analyses of rod ejection cases for the US-APWR first and equilibrium cores.

The applicant calculated the uncertainty between the actual and measured reactor powers for the rod ejection event DCD cases performed using TWINKLE-M (3D) for BOC and EOC conditions and three locations for the ejected rod. The uncertainty evaluation used detector weighting factors calculated separately using the neutron transport code DORT. The analysis indicated that the high power range neutron flux trip occurs when two out of four of the measured reactor power channels [(Proprietary information withheld under 10 CFR 2.390)]. If a single ex-core detector has failed, the trip occurs when the third-highest measured detector power reaches this value. The applicant indicated

that for both the BOC and EOC cases the times when third-highest detector reaches this value are slightly delayed relative to the reactor trip times in the DCD calculations.

To determine the impact of the larger power distribution uncertainty, the applicant performed a sensitivity analysis using a bounding 0.1-second reactor trip time. The applicant concluded that the impact on the overall results is negligible but that the evaluation also indicated that the uncertainty in the power distribution can be larger than that assumed as a part of the analytical limit. In light of this finding, Topical Report MUAP-07010 has been revised to include the explicit calculation of measured power described in this RAI response as shown in Section 5.3, "Spectrum of RCCA Ejection," and Appendix G, "Calculation of High Power Range Neutron Flux Reactor Trip for Rod Ejection Event" of Revision 2. The staff finds that the applicant evaluation adequately addresses the uncertainty in the power range high flux reactor trip setpoint.

To support the applicant's view that no DNB is expected to occur for rod ejection cases where the reactor high flux trip setpoint is approached but not reached, the applicant provided three DNB analysis scenarios as a part of the response to RAI REA-12.2. The analysis results show that during the short-term period the MDNBR remains above the analytical acceptance limit for BOC and EOC conditions for the first-cycle and equilibrium-cycle cores. For the long-term period, the applicant evaluated the effects of other reactor trips (low pressurizer pressure or over temperature ΔT) that occur as a result of declining RCS pressure following a rod ejection event. The sensitivity analyses indicated the maximum number of fuel rods below the MDNBR was bounded by the Chapter 15 analysis assumption. Based on the rod worth uncertainties, conservative assumptions regarding the hot channel factor, the VIPRE-01M inputs and the reduced Doppler reactivity feedback use the staff finds that the applicant's evaluation of low worth rod ejections acceptable.

In summary, the application of the Non-LOCA methodology and the results of the sample calculations of the Spectrum of RCCA Ejection event were reviewed. The staff review focused on the layout, structure and initialization of the TWINKLE-M and VIPRE-01M models, and conservatisms inherent in the methodology, including the magnitude of the reactivity insertion, Doppler feedback, thermal factors affecting calculation of DNB and the calculation of the fuel enthalpy rise for the evaluation of PCMI.

The analysis assumptions in the applicant's RCCA ejection analysis conform to those given in RG 1.77 Appendix A [Ref. 38] and the analysis conservatively ignores the neutron flux rate high trip. The analysis results in a conservative assessment of the fuel centerline temperature, PCMI fuel failure, fuel enthalpy rise, and number of rods in DNB. Therefore, staff finds the Non-LOCA methodology acceptable for evaluating the US-APWR Spectrum of RCCA Ejection event.

5.2.4 Steam System Piping Failure

The applicant presented the specific analysis methodology and a sample analysis for this event in Sections 5.4 and 6.4, respectively, of the topical report.

A break in the US-APWR steam system piping results in an asymmetric cooldown of the RCS because the heat removal by the SG in one loop is much greater than the heat removal by the SGs in the other three loops. The cooldown of the RCS can cause an increase in power due to the negative value of the moderator temperature coefficient. For a large main steam line break (MSLB) the reactivity insertion may be sufficient to cause a return to power after scram has occurred. The post-scram return to power is terminated by the injection of boron into the RCS, by dry-out of the affected steam generator, or a combination of the two.

The applicant's methodology uses MARVEL-M to compute the core power and RCS thermal hydraulic responses.

For HFP initial conditions the DNBR tables within MARVEL-M are used to compute the pre-scram MDNBR. This is acceptable because the core operating conditions are within the range of the independent variables of the DNBR tables. Post-scram MDNBR is calculated using VIPRE-01M. The VIPRE-01M analysis is done as a steady state calculation at certain points at and around the time of the peak core heat flux computed by MARVEL-M. Core pressure and the core inlet enthalpy distribution for VIPRE-01M are obtained from MARVEL-M. In addition, the core power distribution is input to VIPRE-01M. The power distribution is computed by the ANC computer code assuming a stuck rod and the core inlet enthalpy distribution from MARVEL-M at the time of peak power.

The following engineered safeguard features are assumed to be available to mitigate the MSLB:

- Steam line isolation
- Emergency feedwater system (EFWS) isolation
- Main feedwater (MFW) isolation
- Safety injection (SI)

The following RPS trips are available to trip the reactor, initiate safety injection, and isolate systems:

- Low steam line pressure in any loop
- Over-power ΔT high trip
- Over-temperature ΔT high trip
- Pressurizer pressure low trip
- Neutron flux high trip (not credited)
- Containment pressure high trip (not credited)

The following conservatisms are part of the MSLB analysis methodology:

- No credit is taken for the check valves in the main steam lines. All SGs are assumed to blow down until the MSIVs close.
- All EFW is directed to the affected SG
- Failure of one SI train (limiting single failure)
- One SI train out of service for maintenance
- Both available SI trains inject into the unaffected sectors of the core.

As described in topical report Section 5.4 (1), the MSLB event requires the use of several unique models in MARVEL-M. The Moody model [Ref. 56] is used to calculate the break flow; furthermore, only vapor is allowed to flow from the break. This is a conservative assumption because it insures that all the liquid in the SG secondary remains available to boil off and extract heat from the primary side. The flow mixing models at the core inlet and outlet and the weighting of the core sector reactivities take on increased importance. ECCS actuation and the injection of boron play a key role in limiting the post-scram return to power. MARVEL-M's ability to calculate natural circulation is important to those cases which simulate a loss of offsite power.

The sensitivity of the calculated MDNBR to core flow mixing for the HZP MSLB event was illustrated in Appendix E, "Sensitivity Study of the Inlet Mixing Coefficient for Steam System Piping Failure" of the topical report by presenting DNBR as a function of the mixing parameter. RAI 5.4-1 and RAI APP.E-1 [Ref. 19] requested the applicant to substantiate the applicability of the results shown in Appendix E to other AOOs and PAs. In its response [Ref. 19], the applicant stated that the non-uniformity of RCS loop temperatures was greatest for the MSLB event.

Therefore, the sensitivity of DNBR to core flow mixing in other events would be less than that shown in Appendix E. The applicant also asserted that the uniformity of vessel inlet flow mixing was increased with decreasing RCP flow. In particular, it stated that "…*inlet mixing can best be approximated by perfect mixing for very low flows such as during natural circulation conditions*." RAI MSLB-10 asked the applicant to justify this assertion. In its response [Ref. 36], the applicant considered an MSLB with the power lost to the RCPs. In this scenario, the applicant stated, the cold water from the faulted loop would settle to the bottom of the reactor pressure vessel (RPV) lower plenum. In order to reach the core, it would have to rise through the warmer water above and that process would result in nearly perfect mixing. The staff did not agree with the applicant's assessment. A likely scenario for the pumps-off case is for the cold water from the affected loop to settle to the bottom of the RPV and, after displacing the warm water there, be the only water entering the bottom of the core. This situation is similar to no-mixing rather than perfect mixing.

The staff conducted mixing sensitivity runs using MARVEL-M to determine how mixing assumptions affected three key parameters of the MSLB analysis: peak power, affected sector inlet flow at the time of peak power, and affected sector inlet temperature at the time of peak power. The MSLB HZP pumps-off scenario was considered for three core inlet mixing assumptions. The base case used the applicant's mixing assumption. Sensitivity Case 1 assumed perfect mixing, and Sensitivity Case 2 assumed no mixing. The results of the three cases, shown in Table 1, indicate that the base case gives the highest power. The no mixing case gives a slightly smaller peak power, but has a lower inlet temperature and a higher inlet flow than the base case. Thus, the base case would give the lowest MDNBR of the three cases if the case key parameters were transferred to VIPRE-01M.

Table 1 Sensitivity of Key MARVEL-M Parameters to Core Inlet Mixing Assumption for the MSLB with Loss of Offsite Power

Parameter	Base Case	Perfect Mixing (Case 1)	No Mixing (Case 2)
Peak Power (%nom.)	6.9	4.7	6.3
Sector A Inlet Flow, kg/s	398	347	433
Sector A Inlet Temperature, C	178	179	177

Based on the above results, the staff finds that the core inlet mixing assumptions used by the applicant for the HZP pumps-off MSLB case are acceptable. There is reasonable assurance that the MDNBR calculated by the applicant is lower than would be calculated by using other core inlet mixing assumptions.

RAI 5.4-2 asked the applicant to provide the mixing model and substantiate why it is conservative. In its response to RAI 2.1-13 [Ref. 19], the applicant explained where the design and conservative values are given and stated to be based on 1/7 flow mixing experiments

conducted by the applicant and documented in Topical Report MUAP-07022-P, “Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report,” [Ref. 23]. The staff performed an independent calculation of the values of the mixing parameters using Figure 4-19, “Core Inlet Non-dimensional Temperature Distribution” of MUAP-07022-P, and confirmed that the values being used in MARVEL-M are conservative, and therefore acceptable.

RAI 6.4-1 asked if the SLB event is sensitive to the location of the break. In its response [Ref. 19], the applicant stated that the most limiting combination of single failure and break location is with the break located upstream of the main steam line check valve and isolation valves. The presence of the check valve is ignored.

Therefore, all four SGs blow down until MSIV closure, the case analyzed in Section 6.4 of the topical report, resulting in a greater cooldown of the RCS. The applicant’s determination of the limiting break location is reasonable and ignoring the check valve is conservative; therefore, the response is acceptable.

It was noted that the input variable [

(Proprietary information withheld under 10 CFR 2.390)

]. The applicant’s analysis is acceptable. However, while a higher saturation pressure is conservative with respect to reactivity effects, it is non-conservative with respect to MDNBR considerations. Independent calculations indicated that the upper head temperature could drop from 561 K to 555 K (550 °F to 539 °F) in the first 50 seconds of the transient, lowering the upper head saturation pressure from 7.3 MPa to 6.7 MPa (1053 psia to 970 psia). RAI MSLB-1-1 asked the applicant to address the effect that a lower saturation pressure would have on calculated MDNBR.

In its response to RAI MSLB-1-1 [Ref. 43], the applicant performed an MSLB simulation in which with the upper head bypass flow set to [(Proprietary information withheld under 10 CFR 2.390)] lower than the base case. The lower pressure resulted in increased HPSI flow relative to the base case; hence, boron reached the core sooner, terminating the post-scram power increase. The net result was a smaller post-scram peak power and a higher MDNBR relative to the base case. The applicant’s response also provided justification [Ref. 44] for using the W-3 DNBR correlation for pressures below 6.9 MPa (1,000 psia). The response adequately addresses the effect of upper head bypass on the system’s response to an MSLB, and is therefore acceptable.

In its response to RAI MSLB-2 [Ref. 36], the applicant explained that the pump loss coefficient, [(Proprietary information withheld under 10 CFR 2.390)] is used in the analysis of the locked rotor event and provides the pump pressure loss term during natural circulation conditions. While the loss coefficient for forward flow can be obtained from the homologous curves, it is provided as a user input so that a safety margin can be included. The response is

acceptable because it describes a reasonable way to conservatively model pressure losses across the RCP.

RAI MSLB-3 requested the applicant explain why the variables [

(Proprietary information withheld under 10 CFR 2.390)

]
was calculated and why it is conservative, thus providing an acceptable response to RAI MSLB-5 [Ref. 36].

The staff performed an independent calculation of the time that low steam line pressure trip would occur in the SLB event using the lead/lag control system which calculates steam line pressure. [(Proprietary information withheld under 10 CFR 2.390)]

The staff inquired why the steam line pressure trip's lead and lag times were not the same in the MARVEL-M input file and the calculation memo which accompanied the input file. The response to RAI MSLB-4 [Ref. 36] explained that the values in the input file were conservatively biased by [

(Proprietary information withheld under 10 CFR 2.390)

] value is acceptable for the MARVEL-M calculation.

As noted in the previous paragraph, when the reactor returns to power in the SLB analysis nearly all of the power is generated in the fuel assemblies in the affected sector, since that is where the stuck rod is. RAI MSLB-8 noted that the MARVEL-M SLB analysis distributed the

power equally to the 4 quadrants of the core rather than mainly to the affected quadrant, and requested a justification of that assumption. The applicant responded [Ref. 36] that the power is distributed uniformly because non-uniform radial power distributions cannot be utilized in MARVEL-M. It then indicated that so long as the reactivity feedback in MARVEL-M is demonstrated to be conservative the exact details of modeling the power distribution in core are not important. However, the applicant's response was not correct with respect to modeling non-uniform power distributions. [(Proprietary information withheld under 10 CFR 2.390)] successfully used in an independent MARVEL-MSLB calculation by the staff. The staff issued RAI MSLB-7-1 asking the applicant to run an MSLB simulation using a reactivity weighting scheme the staff considered more appropriate.

In its responses to RAI MSLB-7-1 [Ref. 43], and RAI-5 [Ref. 46], the applicant addressed the staff's concerns regarding reactivity weighting. An MSLB simulation was run with both uniform weighting and with almost all the weight given to the affected sector. Both runs were shown to yield nearly the same return to power, with the power for the uniform weighting case being slightly larger.

In its response to RAI-5 [Ref. 46], the applicant also addressed the staff's concern regarding the possibility that the input ANC boron concentration might be too high. The value used in ANC was the core average boron concentration calculated by MARVEL-M. The ANC calculation was redone using the minimum boron calculated by MARVEL-M. A comparison of the reactivity calculated by ANC to that calculated by MARVEL-M showed that the MARVEL-M value was conservatively high. The staff finds that the responses to the foregoing RAIs demonstrate the conservative nature of the MSLB methodology with respect to reactivity weighting and boron concentration in ANC, and are therefore acceptable.

In RAI MSLB-11 the staff asked why fluid temperature increases as one moves from the cold leg to the core inlet. In its response to RAI MSLB-11 [Ref. 36], the applicant noted that at any given time prior to SG dry-out the water at the RCP will be the coldest and this cold front will progress toward the lower plenum. Thus there will always be a temperature gradient between the lower plenum and the SG outlet plenum. The staff concurs with this explanation.

RAI MSLB-12 expressed the concern that the boron front could be smeared numerically with the result that boron would be calculated to arrive at the core sooner than it physically could. In its response [Ref. 36], the applicant stated that the safety injection nozzle in the MARVEL-M analysis was assumed to be in the cold leg whereas it actually is in the downcomer. The applicant claimed that this assumption was adequate to offset the small amount of boron smearing that occurs during the simulation. This response did not allay the staff's concern. The MARVEL-M model assumes that the SI is injected approximately [

(Proprietary information withheld under 10 CFR 2.390)

]. Therefore a better justification is needed of why the boron content in the core is conservative. It is noted that the power rise in the SLB analysis is terminated by boron reaching the core; hence, assurance that the core's boron content is conservatively calculated is important.

The staff issued follow-up RAI MSLB-12-1 which requested the applicant to provide more justification that boron is treated conservatively. In RAI MSLB-20 the staff requested the applicant explain how the ECC line sweep-out volume (volume of unborated water that is in the ECC lines when HPSI begins) is calculated. In its response to MSLB-20 [Ref. 43], the applicant provided the calculation of the sweep-out volume based on the US-APWR design data and a

conservative additive factor. This response showed that the sweep-out volumes used in MARVEL-M were conservative; they were therefore acceptable. In its response to RAI MSLB-12-1 [Ref. 43], the applicant provided a sensitivity calculation - the limiting MSLB case [(Proprietary information withheld under 10 CFR 2.390)]. The peak power for the sensitivity case was not significantly different from the base case peak power. The applicant noted that the time delay for the SI pumps to go from start to full speed is 18 seconds for the offsite power available cases being considered. The MSLB analysis imposes a conservative time delay of 20 seconds for the SI pumps to reach full power. The applicant's responses to MSLB-12-1 and MSLB-20 are acceptable; they provide the justification that boron injection was treated conservatively in the MSLB analyses.

The shutdown reactivity in the SLB analysis is 1.6 percent Δk/k. In its response to RAI MSLB-13 [Ref. 36], the applicant stated that the Mitsubishi reload evaluation methodology (Topical Report MUAP-07026-P, "Mitsubishi Reload Evaluation Methodology,") [Ref. 47] provides procedures for ensuring that this value is less than the shutdown margin for each cycle. If it is not, the SLB event will be re-analyzed. This procedure to ensure that the value used in the present analysis is conservative is acceptable.

The staff was concerned that the SLB methodology did not cover the full range of possible moderator density coefficients (MDCs). RAI MSLB-14 noted that the MDC used in the analysis ranged from [(Proprietary information withheld under 10 CFR 2.390)] (Δk/k)/(gm/cc) at a density of 0.95 gm/cc. The applicant was asked to explain how its analysis supported the value of the moderator density coefficient, 0.51 (Δk/k)/(gm/cc), used in other US-APWR DC Tier 2 Chapter 15 analyses. In its response to MSLB-14 [Ref. 36], the applicant indicated that it did not. In light of this, the staff concluded that the SLB analyses as it is performed only supports a hot full power MDC corresponding to the core loading used to generate the MDC curves used in the present analysis. If a core loading with a higher MDC occurs it must be re-analyzed. In its response to RAI MSLB-14-1 [Ref. 43], the applicant noted that the moderator defect curve used in the MSLB analysis is conservatively calculated by ANC and includes the assumption of a stuck rod. The applicant also noted that the applicant's reload evaluation methodology [Ref. 47] insures that cycle-specific values of key safety parameters, such as the moderator defect curve, are bounded by previous cycle values. If not, the relevant safety analyses are redone. The staff agrees that the moderator feedback is being treated conservatively in the MSLB analysis and finds that appropriate procedures are in place to insure future analyses are done conservatively. Therefore, the responses to RAI MSLB-14 and MSLB-14-1 are acceptable.

In its response to RAI MSLB-15 [Ref. 36], the applicant described the procedure used to develop the power defect reactivity curve. The applicant responded that the curve is developed at [(Proprietary information withheld under 10 CFR 2.390)]. Several power state points are analyzed and the eigenvalue for each is recorded. The defect curve is developed from the eigenvalue information. This procedure is acceptable because it is in accordance with standard engineering practice for developing power defect curves.

The staff issued RAI MSLB-16 to express the concern that the applicant's SLB methodology does not provide for the same moderator density distribution to be used in both VIPRE-01M and ANC. As it is formulated, the methodology calls for a single ANC run and a single VIPRE-01M. The ANC run is made using core inlet boundary conditions from the MARVEL-M run; it outputs a 3-D core power distribution. ANC calculates a core moderator density distribution based on closed channel hydraulics; that is, there is no cross flow between assemblies. The power distribution calculated by ANC is input to VIPRE-01M. VIPRE-01M then calculates a new moderator density distribution based on open channel hydraulics. Thus, it produces a density

distribution that, if it were given to ANC, would produce a power distribution different from what VIPRE-01M is using. A robust methodology should involve iteration between ANC and VIPRE-01M until they converge to a common core power distribution and common moderator density distribution. RAI MSLB-16 requested that the applicant provide justification for using inconsistent power and moderator density distributions in ANC and VIPRE-01M. In its response [Ref. 36], the applicant stated that *"…since ANC uses a closed channel thermal-hydraulic model which does not take into account fluid cross-flow between fuel assemblies, ANC overestimates the fluid density at the boiling region relative to VIPRE-01M, which does account for fluid cross-flow. The ANC power peaking therefore is larger relative to the open channel hydraulic model. As a result, it is more conservative to evaluate DNBRs based on the combination of ANC power distribution and VIPRE-01M fluid density distributions without an iterative process."* The applicant also presented results from an iterative calculation using ANC and VIPRE-01M and stated that the calculation showed that the converged solution gives a MDNBR which is greater than that given by the initial calculation. The results do not provide adequate information to support the applicant's position. More importantly, even if the hypothesis advanced by the response is true, it does not address the issue of whether the local power factors are converged. MDNBR is not the only SAFDL that should be evaluated; the fuel centerline melt (FCM) SAFDL must be evaluated.

That evaluation requires that the local power distribution be accurately determined. That determination can only be assured by insuring convergent power and density distributions in ANC and VIPRE-01M. RAI MSLB-16-1 and RAI-6 requested for additional information regarding the iterative study done by the applicant.

In its response to MSLB-16-1 [Ref. 43], the applicant supplied the axial power profile, F_q, location of F_q, axial void distribution, and MDNBR for each of [(Proprietary information withheld under 10 CFR 2.390)] than the converged MDNBR value. In its response to RAI-6 [Ref. 46], the applicant provided a detailed explanation of the data transfers between VIPRE-01M and ANC. Based on these responses, the staff finds that there is reasonable assurance that the MDNBR and F_q values calculated by the applicant's methodology, with no iteration between VIPRE-01M and ANC, is conservative. Therefore the staff finds the data transfer procedure acceptable and RAIs MSLB-16, MSLB-16-1, and RAI-6 are resolved.

A complete transient evaluation methodology requires the evaluation of all relevant SAFDLs, not just MDNBR. Therefore, RAI MSLB-18 requested the applicant provide an evaluation of the fuel centerline temperature relative to its SAFDL and to address the possibility of fuel failure due to PCMI. In its response to RAI MSLB-18 [Ref. 43], the applicant provided the requested analyses. The fuel centerline temperature analysis showed that the calculated centerline temperature was far below the SAFDL value for the limiting MSLB case. The PCMI analysis showed the prompt fuel enthalpy [(Proprietary information withheld under 10 CFR 2.390)] MSLB case, even when very conservative assumptions were made for reactivity insertion rates and local peaking factors. Because of the large margin between the calculated fuel centerline temperature and prompt fuel enthalpy rise and their respective SAFDL limits, the staff finds that the applicant's MSLB methodology does not need to be supplemented to include the evaluation of these two safety parameters.

In RAI MSLB-19 [Ref. 43], the staff asked which DNBR correlation was used for the MSLB, LOOP analysis (i.e., low pressure and flows) and was the correlation within its applicable range during LOOP plant event. In its response to RAI MSLB-19, the applicant stated that the W-3

DNB correlation was used for the MSLB HZP LOOP case. Information to support the use of the W-3 correlation for the low mass flux calculated for this MSLB case was also presented. This supporting information demonstrated that the W-3 correlation exhibited no anomalous behavior when it was compared to CHF data with mass fluxes below the lower limit of 1360 kg/m^2-s ($1x10^6$ $lbm/hr\text{-}ft^2$) of the W-3 correlation. The applicant also noted that the HZP pumps-off case is bounded by the HZP pumps-on case, having a peak power which is less than half of what is calculated for the limiting MSLB case. The low heat flux and high subcooling calculated for the HZP pumps-off case make the occurrence of DNB unlikely. The staff finds the use of the W-3 DNB correlation acceptable for the MSLB HZP pumps-off case. The applicant has adequately demonstrated that the correlation gives a reasonable estimate of the DNBR for the low flow condition observed in that case and the response to RAI MSLB-19 is acceptable.

In summary, the staff reviewed the application of the Non-LOCA methodology and the results of the sample calculations of the Steam System Piping Failure event. The staff's review focused on fluid mixing in the RV, core power distribution, reactivity weighting and feedback, boron injection and transport, and DNB modeling. For the reasons discussed in the above paragraphs, the staff finds the applicant's revised Non-LOCA methodology acceptable for evaluating the US-APWR Steam System Failure event.

5.2.5 Feedwater System Pipe Break

The applicant presented the specific analysis methodology and a sample analysis for this event in Sections 5.5 and 6.5, respectively, of the topical report.

The Feedwater Line Break (FWLB) event is a postulated accident and is defined as a major break in a feedwater pipe. Like the SLB, it induces asymmetric thermal hydraulic conditions into the RCS. The FWLB event can be a heat-up event, a cool-down event, or a combination of both. The FWLB as a cooldown event is bounded by the SLB event; therefore, the FWLB is analyzed as a heat-up event and analysis conditions are biased appropriately. The SLB event bounds the FWLB cooldown event because it retains liquid inventory in the SG secondary side longer.

The limiting FWLB heat-up event is a double-ended rupture between the SG feedwater nozzle and the last feedwater line check valve.

The following RPS trips are available to automatically trip the reactor and help mitigate the FWLB event:

- SG water level low trip in any loop
- Pressurizer pressure high trip (not credited)
- Pressurizer level high trip (not credited)

The following engineered safeguard features are assumed to be available to mitigate an FWLB:

- EFWS actuation, automatic
- EFWS isolation
- SI

The FWLB is modeled in MARVEL-M using the assumption that the main feedwater flow stops at the initiation of the transient. This results initially in a slow heat-up of the RCS. When the low

SG level setpoint is reached, the reactor trip signal is generated. Simultaneously, a double-ended guillotine break of the feedwater line and LOOP are assumed to occur. Break flow is assumed to be only saturated water and is computed using the Moody model [Ref 56]. These two assumptions maximize the removal of liquid from the SG and cause a conservatively-rapid loss of SG heat removal capability for the affected SG. EFW is assumed not to be available because it is automatically isolated due to a low SG pressure signal. The check valve in the affected SG steam line isolates the affected SG from the rest of the steam system.

The FWLB analysis is used to evaluate the peak RCS pressure; therefore, RCS initial conditions (core power, initial RCS pressure and temperature) and core parameters (reactivity feedback coefficients, decay heat) are selected to maximize RCS pressure. The staff reviewed the input for the FWLB sample problem and confirmed that the initial conditions listed were biased to maximize RCS pressure.

RAI 6.5-1 questioned the procedure of performing the FWLB analysis using the assumption that the main feedwater flow stops at the initiation of the event (which leads to an occurrence of an SG low level trip). It asked if the pressurizer water volume was sensitive to perturbations in this assumption. In its response [Ref. 19], the applicant stated that the procedure used is conservative in that it reduces the SG inventory available to cool down the primary system when the break opens and results in a slow heat up of the primary prior to opening of the break. This latter feature causes an in-surge into the pressurizer. Therefore, the procedure is conservative relative to maximizing the pressurizer liquid volume during the event.

RAI FWLB-1 asked if a break spectrum study had been completed. In its response to RAI FWLB-1 [Ref. 36], the applicant stated that a break spectrum had not been conducted because smaller breaks are bounded by the case presented, which is a double-ended rupture of the feedwater line. Smaller breaks would result in a slower reduction in the affected steam generator's heat removal capability and therefore less of a heat up of the primary system. The applicant's disposition of smaller breaks is reasonable.

The applicant's FWLB analysis assumes LOOP occurs coincident with the break occurrence. RAI FWLB-2 requested more information regarding the sensitivity of the analysis to LOOP timing. In its response [Ref. 36], the applicant stated that the requested information had already been submitted in its response to Question 15.0.0-3 of RAI 297-2287 [Ref. 48], and showed that assuming a minimum time delay resulted in the most severe results. The relevant RAI response was evaluated during the Chapter 15 DCD FWLB analysis and found acceptable.

In RAI 5.5-1, the applicant was asked if the FWLB transient analysis was sensitive to the timing of the transition to natural circulation and if the core flow mixing factors were the same for the forced flow and natural circulation regimes. In its response [Ref. 19], the applicant stated that there is no explicit model to transition from forced flow to natural circulation. As the RCPs coast down, the gravity head terms in the MARVEL-M flow equations increase in importance relative to the RCP head terms. Thus, the transition to natural circulation is built into the MARVEL-M numerical solution. With respect to mixing factors, the same mixing factors (FMXI and FMXO) are used for all flow conditions. The applicant asserted that the mixing in the upper and lower plena would be near perfect during natural circulation conditions. In its response to RAI 5.5-2 [Ref. 21], the applicant provided a sensitivity study to examine the effect of the mixing factor values on the FWLB results. Two sensitivity calculations were conducted. One had the inlet mixing factor value decreased by 10 percent; the second had the inlet mixing factor value increased by 10 percent. The applicant compared the results from the sensitivity calculations to

the base case simulation. The comparison showed that the ±10 percent variation in the inlet mixing factor value had a negligible effect upon the RCS pressure response and the pressurizer water level response. The response is acceptable because it demonstrates the insensitivity of the key safety parameters to modest changes in the value of the inlet mixing factor used in the FWLB analysis.

The methodology for FWLB assumes that EFW is supplied to only two of the four SGs[1]. Therefore, after the RCPs coast down, two of the loops (1 and 2) will stagnate and the other two (3 and 4) will remove core heat via natural circulation. Under these circumstances there will be little mixing between the fluids in the flowing and stagnant loops. The core inlet temperature will be nearly the same as the cold leg temperature of the flowing loops. An examination of the MARVEL-M simulation shows that the loop flows in all loops are nearly the same throughout the FWLB simulation. The difference between the hot and cold leg temperatures in the affected loop is zero for times greater than 1500 s, while the difference between the hot and cold leg temperatures in Loop 4 is [(Proprietary information withheld under 10 CFR 2.390)] percent of nominal. There is no apparent driving force for the flow in Loop 1.

The applicant selected the failure of an EFWS as the limiting single failure for the FWLB analysis. RAI FWLB-4 asked why the possibility of one EFWS being out of service for maintenance was not considered in the analysis. In its response [Ref. 43], the applicant explained that the US-APWR Technical Specifications (TS) require the EFW systems to be cross-tied if any one system is out for maintenance. Consequently, assuming a system out of service means that EFW is available to all SGs. Assuming all systems are in service means that the systems are not cross-tied. In this case, one EFW is lost due to the break and another is lost to an assumed single failure. EFW is therefore available to only two of the SGs. This is a more limiting condition than EFW available to all the SGs. Therefore the failure of an EFWS as the limiting single failure assumption for FWLB analysis is acceptable.

The MARVEL-M FWLB simulation shows that all four reactor loops have nearly the same flow after the RCPs have coasted down. This appears to be non-physical behavior since only two of the loops have primary-to-secondary heat transfer. One expects natural circulation to be established in these two loops only – flows in the other two loops should stagnate. RAI FWLB-5 requested an explanation of the MARVEL-M calculated loop flows. In its response [Ref. 43], the applicant stated that some flow would be expected in each of the four reactor loops because of the density difference between the fluid in the RPV downcomer and the core. It further stated that the loop flows would be greater in the loops with SG heat transfer if the length of the SG tubes were considered. [

(Proprietary information withheld under 10 CFR 2.390)

]. This is acceptable because the FWLB analysis is conservative with respect to pressurizer overfill.

[1] Automatic EFW isolation in the affected SG (Loop 1) and EFW lost to one intact SG (Loop 2) due to single failure.

In summary, the staff reviewed the application of the Non-LOCA methodology and the results of the sample calculations of the Feedwater System Pipe Break event. The staff's review focused on trip setpoints, systems assumed available and unavailable, break modeling and fluid mixing. The staff finds the Non-LOCA methodology acceptable for evaluating the US-APWR Feedwater System Pipe Failure event.

5.2.6 Steam Generator Tube Rupture

The applicant presented the specific analysis methodology and a sample analysis for this event in Sections 5.6, "Steam Generator Tube Rupture," and 6.6, "Steam Generator Tube Rupture," respectively, of the topical report.

The SG tube rupture (SGTR) event is postulated as the complete severance of one SG tube. The rupture causes reactor coolant to flow into the SG secondary side. The largest break flow occurs when the rupture occurs at the tube sheet on the cold leg side of the SG tube. Since the primary system water may be radioactive, the event may cause a radiation release to the steam system and potentially to the atmosphere. Radiation detectors in the steam lines alert the operator to the occurrence of a SGTR. The operator may manually scram the reactor, but a substantial period of time, usually at least 30 minutes, is assumed to pass before this action is credited.

The following signals are available to automatically trip the reactor:

- Over temperature delta-T high trip (not credited)
- Pressurizer pressure low trip
- SG water level high-high trip

The following engineered safeguard features are assumed to be available to mitigate a SGTR:

- EFWS actuation
- EFWS isolation
- SI

The SGTR event ultimately is mitigated by operator action. The operator identifies and isolates the affected SG by closing its main steam and main feedwater isolation valves. Next the operator terminates the leak flow by equalizing the RCS and affected SG secondary pressures and terminating ECCS flow. The pressure equalization is accomplished by reducing the RCS temperature (and hence the RCS pressure) through cycling the intact SG relief valves and by reducing the RCS pressure through cycling the pressurizer relief valves.

Two cases are analyzed, one to evaluate the maximum steam release to the atmosphere and a second to evaluate SG overfill. In the steam release case the EFWS train to the affected SG is assumed to fail (limiting single failure) because that increases the steam release through the affected SG safety and relief valves. Additionally, a secondary relief valve is assumed to stick open after it automatically opens. The steam release through this stuck open valve is assumed to be terminated by automatic closure of the associated block valve.

For the SG overfill case the main feedwater is isolated by the high-high steam generator water level trip signal and EFW is assumed to remain available to the affected SG for a conservative length of time.

The MARVEL-M computer code is used to simulate the plant transient response to a SGTR from inception to the time of RCS/SG secondary pressure equalization. The staff evaluation therefore is focused on the ability of MARVEL-M to adequately perform that analysis. Two specific MARVEL model features, important to the SGTR simulation, are the SG tube break flow model and the model for simulating flashing in the upper head of the reactor pressure vessel.

In Appendix F of the Topical report, the applicant claims that the break flow rate to the secondary side is conservatively calculated by first calculating the break flow based upon the initial pressures in the primary and secondary systems. The transient break flow is then computed by multiplying the initial break flow by the square root of the ratio of the transient primary-to-secondary pressure difference to the nominal primary-to-secondary pressure difference. The initial break flow is calculated using the Zaloudek correlation [Ref. 49] with C=1.0 instead of 0.95. The Zaloudek correlation is

$$G = C\sqrt{2g\rho(p_{up} - p_{sat})},$$

where G is the critical mass flux, C is a discharge coefficient, 0.95, g is the gravitational constant, ρ is the saturated liquid density, P_{up} is the upstream pressure, and P_{sat} is the saturation pressure. Several RAIs were submitted to determine exactly how the initial critical flow is calculated for the SGTR event.

In its response to RAI App.F-1 [Ref. 19], the applicant presented the equation used to compute the initial break flow. A modified version of the Zaloudek correlation was used, not the original Zaloudek correlation. Zaloudek's correlation uses the upstream stagnation pressure as a key parameter. The applicant's modification replaces the upstream stagnation pressure by the pressure at the break. The applicant provided no justification for its modification of Zaloudek's correlation. Therefore, RAI App.F-1-1 [Ref. 22] asked the applicant to provide validation for the "modified Zaloudek" correlation. In its response to RAI App.F-1-1 [Ref. 22], the applicant noted that it had compared MARVEL-M to an actual SGTR event as a MARVEL-M validation, described in response to RAI 3.1-6. In order to get MARVEL-M to accurately simulate the measured [(Proprietary information withheld under 10 CFR 2.390)] for the MARVEL-M validation. Since it is using a value of 1.0 in its US-APWR SGTR analysis, the applicant stated that the break flow for that analysis is conservative. The staff did not agree as a smaller value of the discharge coefficient may have been needed in the plant SGTR simulation because of break geometry (i.e., not a double-ended break). A discharge coefficient of 1.0 would be appropriate for the double-ended break SGTR licensing analysis and hence the applicant had not demonstrated its break flow calculation is conservative.

In its response to RAI SGTR-6 [Ref. 43], the applicant attempted to justify using the pressure at the break instead of the upstream pressure in the Zaloudek correlation. The applicant cited experimental data showing that critical flow decreases with an increase in the length-to-diameter ratio (L/D) of the discharge tube. The staff agreed with this observation but did not agree with the applicant's contention that this observation justifies the use of the break pressure instead of the upstream pressure for the Zaloudek correlation.

The staff conducted independent calculations of the SGTR event using RELAP5/MOD3.3. It also reviewed SGTR simulations for operating plants. The staff's investigation showed total calculated initial break flows lie between 18 and 29 kg/s (40 and 66 lbm/s) for the SGTR event. The 29 kg/s value corresponds to the rupture of a tube with a flow area 1.36 times larger than a US-APWR SG tube. Dividing the 29 kg/s value by 1.36 yields 21.3 kg/s (47.0 lbm/s). The staff therefore concludes that the applicant's value of 25 kg/s (55 lbm/s) for the initial break flow in the SGTR event for the US-APWR is reasonable and acceptable.

In Appendix F, "Detailed Break Flow Model for Steam Generator Tube Rupture" of the Topical report, a comparison of the integrated break flow from the above equation is compared to what is called a realistic break flow calculation. The realistic calculation accounts for the pressure loss along SG tube from the hot side to the break. It is shown that the break flow as calculated in MARVEL-M is always greater than the realistic break flow.

If the pressure in the primary system drops sufficiently and the inventory loss is sufficient to empty the pressurizer, the RV upper head may flash. RAI 5.6-1 asked if upper head flashing could affect RV upper or lower plenum mixing. In its response [Ref. 19], the applicant stated that no upper head flashing occurred in the SGTR simulation.

The applicant further explained that if flashing were to occur in the upper head it would have no impact on MARVEL-M upper plenum mixing calculations because the flow from the upper plenum would flow equally to each of the four upper plenum nodes. Lower plenum mixing would also be unaffected since upper plenum flashing would only increase loop flows and the lower plenum mixing model is independent of loop flows. The staff agrees with the applicant's assessment of the effect of RV upper head flashing on the fluid mixing calculated for the upper and lower plena.

The SGTR methodology assumes manual reactor trip at 900 seconds. RAI 6.6-1 asked for the basis for choosing 900 seconds and if the transient results would be different if the time of trip were changed. In its response [Ref. 19], the applicant responded that in the event of an SGTR the PCMS [plant control and monitoring system]-based N-16 radiation alarm will sound within two minutes and prompt the operator to trip the reactor. Therefore, the 900 second time being assumed in the analysis is a significantly conservative assumption. The applicant stated that the response time assumed will be verified to be conservative during operator training. The staff concludes that an assumption of 900 seconds for the operator to initiate reactor trip is conservative.

In summary, the staff reviewed the application of the Non-LOCA methodology and the results of the sample calculations of the SGTR event. The staff's review focused on the tube rupture break flow model, RV upper head flashing and fluid mixing and the timing of the assumed operator action to manually trip the reactor. The staff finds the applicant's Non-LOCA methodology acceptable for evaluating the US-APWR SGTR event.

5.3 Independent Analyses

The staff conducted Non-LOCA simulations for the US-APWR using both the RELAP5/MOD3.3 and MARVEL-M codes. These simulations provided independent verification that analysis conservatisms being claimed by the applicant were indeed conservative. To support these independent analyses, the applicant provided the MARVEL-M input files, instructions for obtaining plotted output from MARVEL-M, and a detailed summary of the SLB, loss of load (LOL) and SGTR input files in its responses to RAI-1, RAI-2, RAI-3, and RAI-4 [Ref. 50].

5.3.1 RELAP5/MOD3.3 Simulations

The following Chapter-15 Non-LOCA events were simulated using RELAP5/MOD3.3 (R5M33) and are documented in Technical Report ISL-NSAO-TR-10-01, "US-APWR Non-LOCA RELAP5/MOD3.3 Confirmatory Runs," [Ref. 25]:

- LOL, US-APWR DCD Tier 2 Section 15.2.1, minimum departure from nucleate boiling ratio (MDNBR) and primary system overpressure cases
- SGTR, US-APWR DCD Tier 2 Section 15.6.3, maximum radiological release case
- MSLB, US-APWR DCD Tier 2 Section 15.1.5, HZP with offsite power available.

The LOL cases and the SGTR case used a RELAP5/MOD3.3 base model that independently represents the four US-APWR reactor coolant loops. The RV is modeled with a single downcomer flow channel, and the lower plenum, upper plenum, and upper head. The core model consists of two regions: one representing a hot assembly, and the other representing the remaining assemblies. For the MSLB simulations the RELAP5/MOD3.3 base model was modified so that each coolant loop had its own corresponding RV quadrant. Mixing between the quadrants at the upper plenum and lower plenum was simulated using time-dependent junctions and the RELAP5/MOD3.3 control system model. The degree of mixing between the affected loop and the other loops was specified via user input mixing factors and could be varied from no mixing to perfect mixing. For comparisons to MARVEL-M, the mixing factors in the RELAP5/MOD3.3 model were specified to be the same as in the MARVEL-M model.

A more complete description of the RELAP5/MOD3.3 US-APWR models can be found in [Ref. 25].

<u>LOL Confirmatory Analysis</u>

For the LOL event, the RELAP5/MOD3.3 LOL confirmatory runs showed good agreement with the MARVEL-M results presented in the US-APWR DCD. The MDNBR calculated by MARVEL-M was lower than that calculated by RELAP5/MOD3.3, as shown in Figure 1 below. The peak RCS pressure calculated by MARVEL-M was higher than that calculated by RELAP5/MOD3.3, as shown in Figure 2 below. The timing of events and the transient responses of key variables with RELAP5 were very similar to those calculated with MARVEL-M, as shown in Table 2 below. The RELAP5/MOD3.3 LOL confirmatory runs show that MARVEL-M results in the US-APWR DCD Tier 2 are conservative.

Table 2 Sequence of Events for Loss of Load MDNBR Case.

Event	**DCD**	**R5M33**
	Time (s)	
Turbine trip, loss of main feedwater flow.	0.0	0.0
High pressurizer pressure signal.	6.7	6.3
Reactor trip initiated.	8.5	8.0
Pressurizer safety relief valves open.	8.6	8.6
MDNBR occurs.	9.5	0.0*
Main steam safety valves open.	9.7	10.0

Event	DCD	R5M33
	Time (s)	
Peak RCP outlet pressure occurs.	10.3	10.3
Peak main steam system pressure occurs.	14.3	15.0

* MDNBR of 2.013 occurs at time 0.0 s. Secondary MDNBR of 2.082 occurs at 9.7 s.

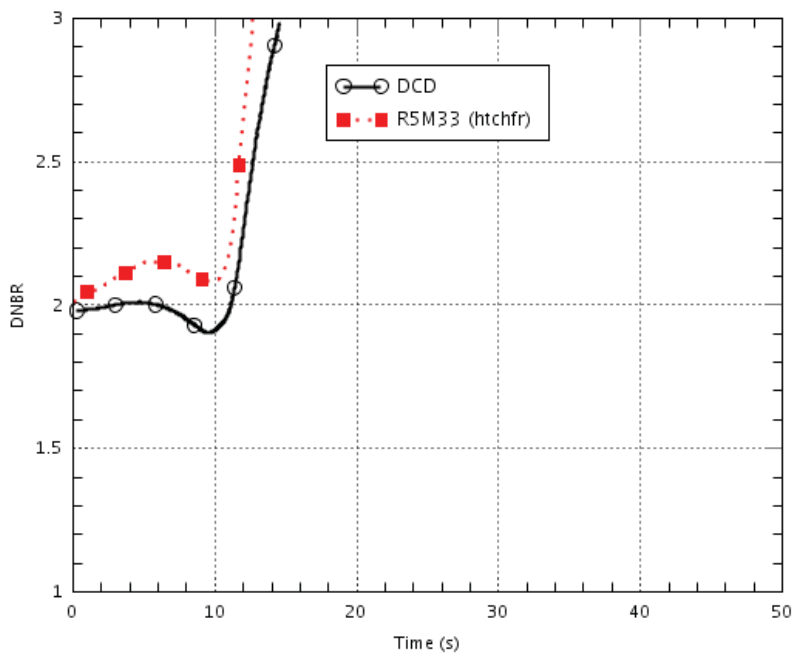

Figure 1 MDNBR for the LOL Cases

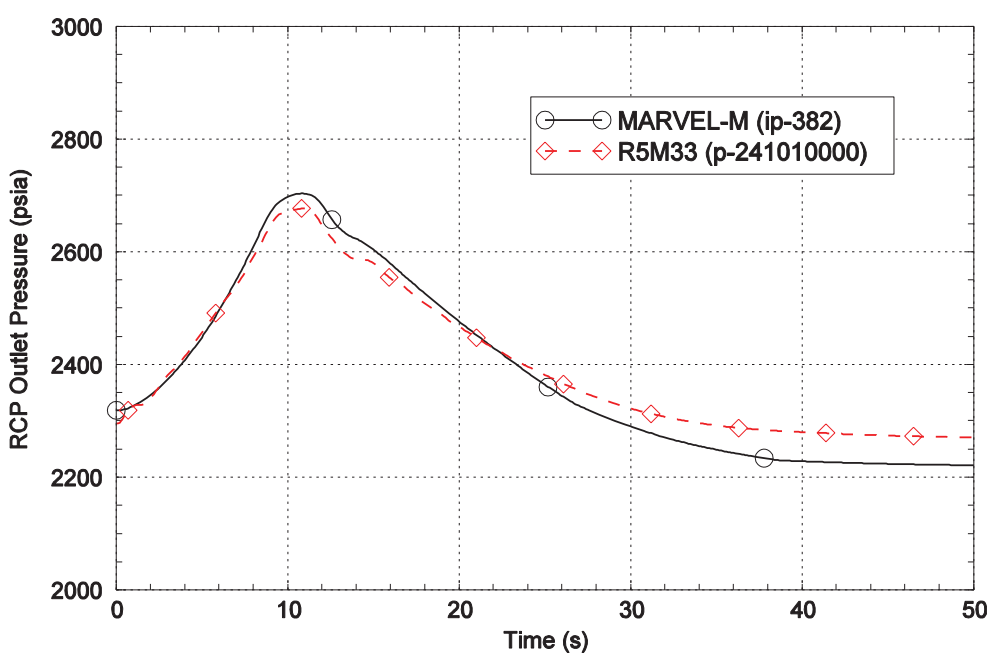

Figure 2 Peak Pressure for the LOL Overpressure Cases

SGTR Confirmatory Analysis

For the SGTR event comparison to MARVEL-M, the RELAP5/MOD3.3 SGTR simulation's break flow was adjusted to be in reasonable agreement with that calculated by MARVEL-M. Assumed operator actions and equipment failures were identical in the two simulations.
Thus, the goal of the confirmatory calculation was to see if MARVEL-M and RELAP5/MOD3.3 would give similar results when given the same initial and boundary conditions. Table 3 and Figure 3 through Figure 5 below show that the two codes provide similar results.

Regarding the results of the RELAP5/MOD3.3 SGTR confirmatory calculations, it is stated in the Staff's confirmatory analysis, ISL-NSAO-TR-10-01 [Ref. 25], that "*Considering the complexity of the SGTR radiological dose evaluation case event scenario and the significant differences between the plant modeling approaches and computer codes employed, the R5M33 and DCD calculations agree remarkably well*." Based on this assessment it is reasonable to conclude MARVEL-M is capable of simulating the SGTR event.

Table 3 Calculated Sequences of Events for SGTR Radiological Dose Evaluation Case

Event	MARVEL-M	R5M33
	Time (s)	
Double-ended rupture of single tube in SG B.	0	0
Manual reactor trip. Scram rod insertion begins. Loss of offsite power assumed.	900	900
Reactor coolant pump power tripped.	900	900
Main feedwater isolation.	900	900
Turbine trip.	900	900
Affected SG B isolated (MSIV closed).	1,200	1,200
Initiation of RCS cooling by the operator. Operator opens MSDVs on intact SG A and intact SG C.	1,500	1,500
MSRV on affected SG B assumed to stick open.	1,500	1,500
Safety injection actuation signal. CVCS and letdown flows terminated. ECCS (HPI) flow initiated.	1,634	1,648
EFW flows initiated to intact SG A and intact SG C.	1,774	1,788
Operator assumed to isolate the flow through the stuck-open MSRV on affected SG B by closing the block valve.	1,826	1,820
Operator assumed to open the pressurizer SDV to depressurize the RCS.	2,717	2,717
RCS subcooling exceeds 50°F, operator closes MSDVs on intact SG A and intact SG C.	2,819	2,816
Operator assumed to close the pressurizer SDV.	2,848	2,848
Operator assumed to terminate ECCS flow.	2,880	2,880
RCS subcooling falls below 50°F, operator reopens MSDVs on intact SG A and intact SG C and they remain open thereafter.	3,117	3,000
Flow through the ruptured tube in SG B terminated	4,183	6,811
Calculation ended.	6,000	8,000

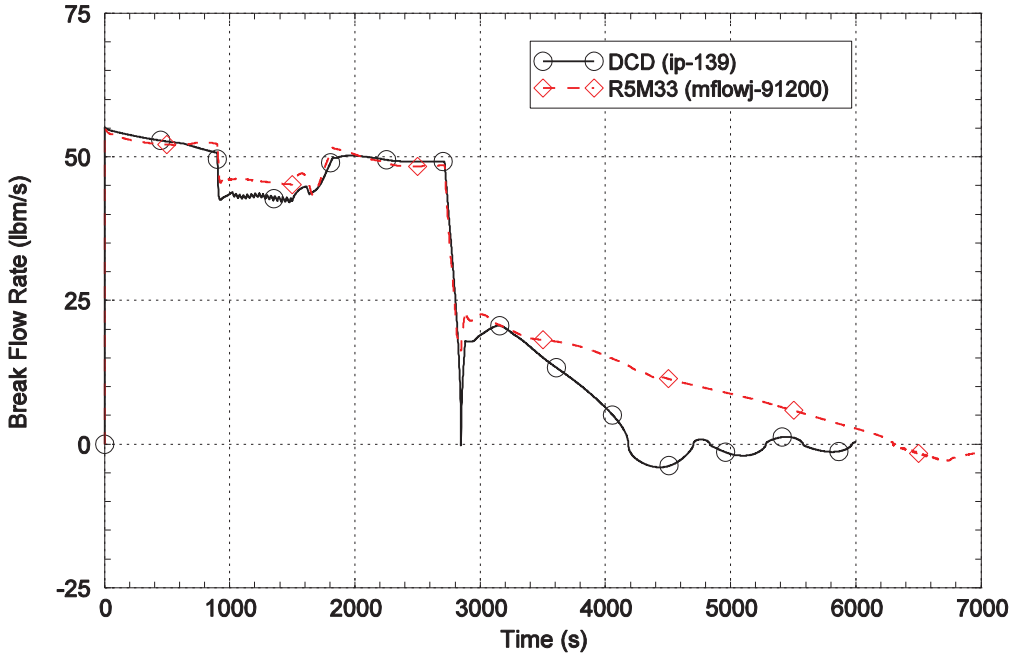

Figure 3 SGTR Event - Break Flow

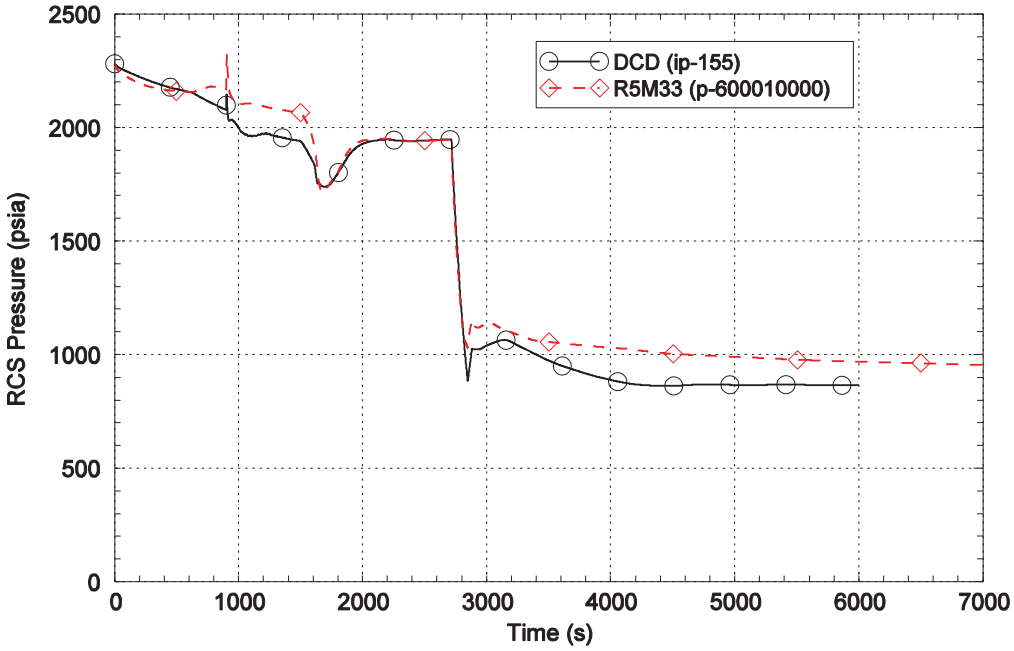

Figure 4 SGTR Event - Pressurizer Pressure

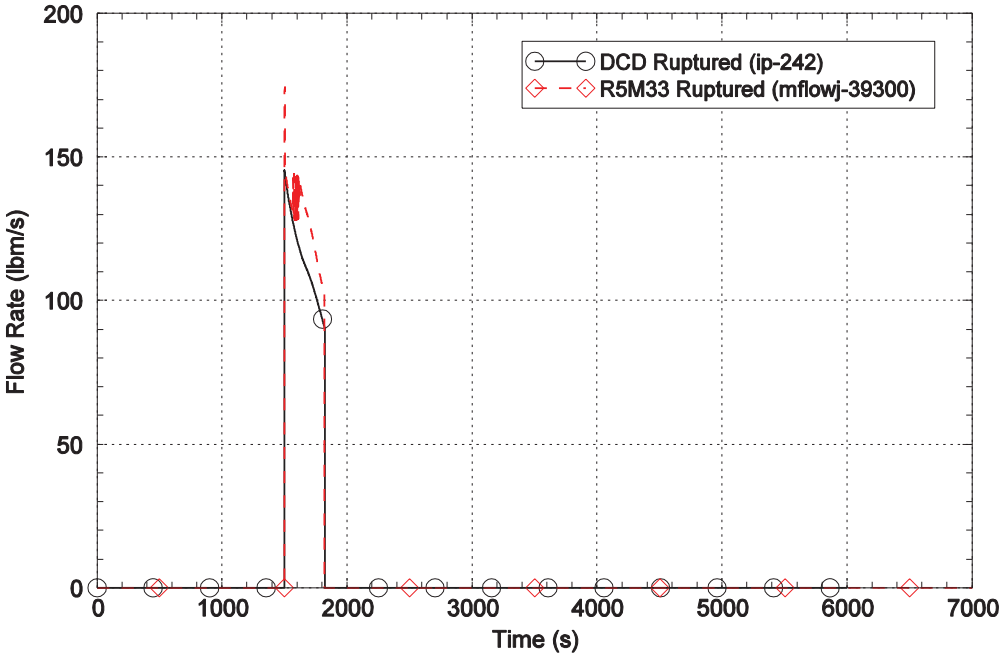

Figure 5 SGTR Event - Safety Valve Flow in Affected SG

MSLB Confirmatory Analysis

The RELAP5/MOD3.3 simulation of the HZP MSLB with RCPs running agreed closely with the MARVEL-M result. The main difference was that RELAP5/MOD3.3 calculated a slightly slower RCS pressure decline, as shown in Figure 6 below. This, in turn, caused SI injection to occur slightly later in the RELAP5/MOD3.3 calculation. In this MSLB event, the rise in core power is terminated by boron reaching the core; therefore, a delay in SI injection leads to a higher return to power. Whereas the US-APWR DCD Tier 2 calculation shows [(Proprietary information withheld under 10 CFR 2.390)], as shown in Figure 7 below. Technical Report ISL-NSAO-TR-10-01 [Ref. 25] does not identify a reason for the slight difference in the RCS pressure response with the two codes. A likely cause is that different models are used to compute the pressure in the pressurizer during a flow out-surge. This analysis demonstrates that the RELAP5/MOD3.3 MSLB confirmatory calculation is in good agreement with the corresponding MARVEL-M calculation in the DCD.

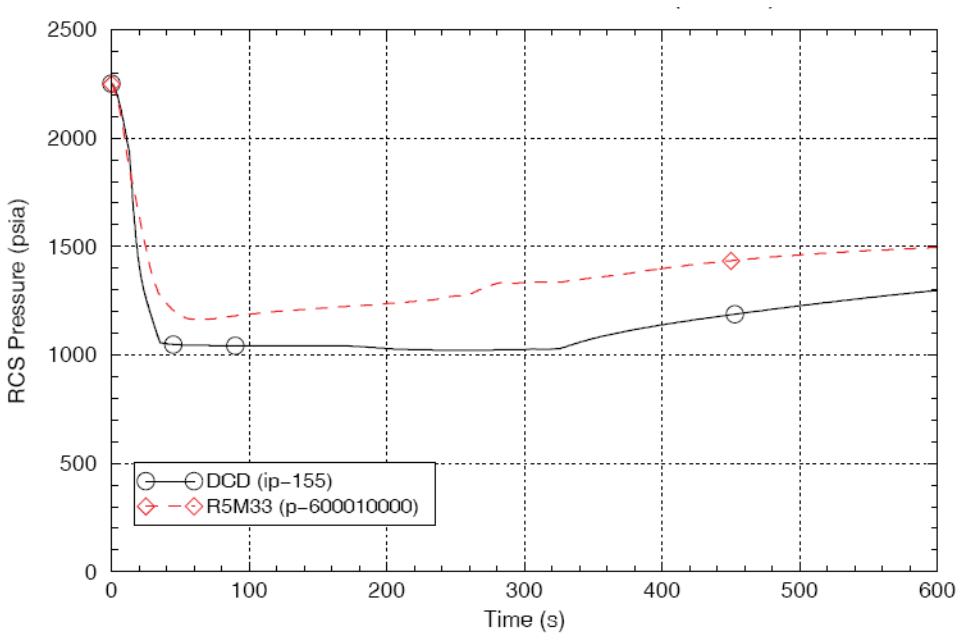

Figure 6 MSLB Event - RCS Pressure

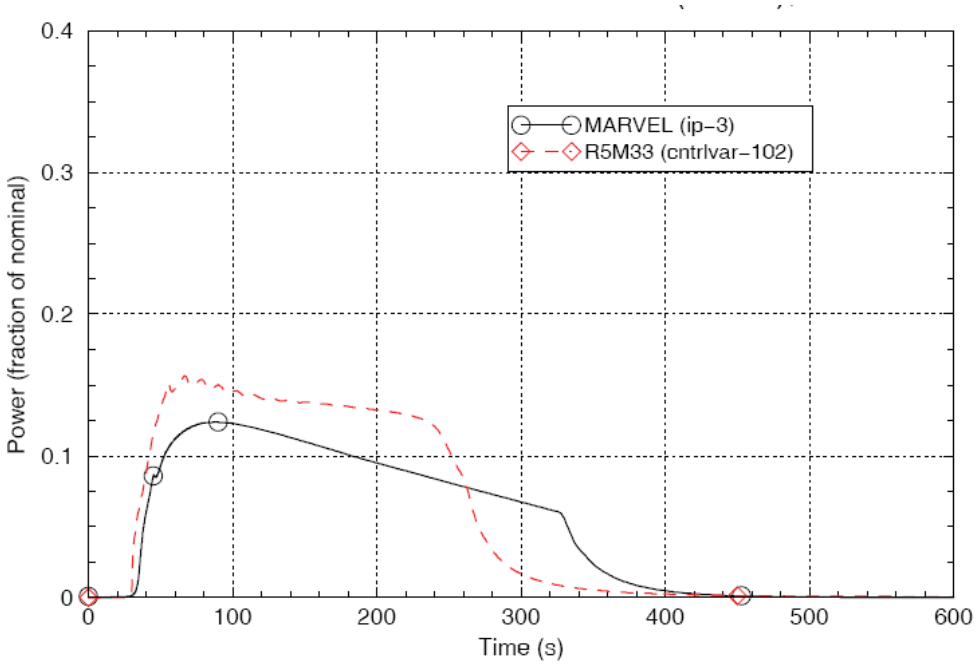

Figure 7 MSLB Event - Normalized Reactor Power

5.3.2 MARVEL-M Simulations

The staff also performed many independent MARVEL-M simulations using the code executable and DCD Chapter 15 analysis input files supplied by the applicant. These in-house calculations allowed the staff to examine the DCD analyses in much more detail than simply reviewing the information provided in the US-APWR DCD and the Non-LOCA methodology Topical report. It also assisted in the understanding of the input and modeling capabilities of MARVEL-M. These independent MARVEL-M calculations were an important part of the staff's review.

Complete Loss of Flow

Many MARVEL-M simulations of the CLOF event were conducted to perform a detailed examination of the code's conservation of mass and energy. Based on the results of the CLOF simulations, it was concluded that mass and energy were conserved in MARVEL-M.

FWLB

The US-APWR DCD FWLB simulation [

(Proprietary information withheld under 10 CFR 2.390)

] of conservatism in the calculated pressurizer liquid volume.

(Figure 8 - Proprietary information withheld under 10 CFR 2.390)

Figure 8 FWLB Event - Liquid Volume in the Pressurizer

5.4 Appendix A - Review of the MARVEL-M computer code

A.0 Introduction

[(Proprietary information withheld under 10 CFR 2.390)

]

A.1 Neutron Kinetics Equation

A.1.1 Decay Heat and Fission Power

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.1.2 Reactivity Coefficient

[(Proprietary information withheld under 10 CFR 2.390)

]

[

(Proprietary information withheld under 10 CFR 2.390)

]

A.2 Fuel Rod and Core Thermal Kinetics Equation

A.2.1 Fuel Thermal Model

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.2.2 Reactor Core Flow Model and Core Power Distribution

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.2.3 Hot Channel Fuel Thermal Model

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.2.4 DNBR Evaluation Model

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3 Equations for Reactor Coolant Flow Sections

[(Proprietary information withheld under 10 CFR 2.390)

]

[

(Proprietary information withheld under 10 CFR 2.390)

]

A.3 1 Reactor Core Coolant Thermal Kinetics Equation (HEAT)

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.2 Mixing Plenum Equation (MIXG)

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.3 Transport in Pipe (SLUG)

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.4 SG Primary Side Equation (HEEX)

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.5 Dead Volume (Reactor Vessel Head Volume) (MIXD)

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.6 Reactor Vessel Upper Plenum (MIXS)

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.7 Pressure Gradient in Reactor Coolant System

[(Proprietary information withheld under 10 CFR 2.390)

]

A.3.8 Thick Metal Effect

[(Proprietary information withheld under 10 CFR 2.390)

]

A.4 Mixing and Flow Model in Reactor Vessel

[(Proprietary information withheld under 10 CFR 2.390)

]

A.4.1 Mixing in the Downcomer, Reactor Vessel Lower Plenum and Upper Plenum

[(Proprietary information withheld under 10 CFR 2.390)

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[(Proprietary information withheld under 10 CFR 2.390)

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[(Proprietary information withheld under 10 CFR 2.390)

]

A.4.2 Core Bypass Flow

[(Proprietary information withheld under 10 CFR 2.390)

]

A.5 SG Model

[(Proprietary information withheld under 10 CFR 2.390)

]

A.5.1 Heat Transfer Coefficient

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.5.2 SG Water Level

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.5.3 Heat Transfer Area

[(Proprietary information withheld under 10 CFR 2.390)

]

A.5.4 SG Secondary-Side Thermal Kinetics Equation

[(Proprietary information withheld under 10 CFR 2.390)

]

A.5.5 Other Special Models

[(Proprietary information withheld under 10 CFR 2.390)

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[(Proprietary information withheld under 10 CFR 2.390)

]

A.6 Pressurizer

[(Proprietary information withheld under 10 CFR 2.390)

]

A.6.1 Water Phase Model

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.6.2 Steam Phase Model

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.6.3 Pressurizer Pressure Control

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.7 Transient Flow Performance

[(Proprietary information withheld under 10 CFR 2.390)

]

A.7.1 Reactor Coolant Pump Models and Flow Transient

[(Proprietary information withheld under 10 CFR 2.390)

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[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)
]

A.7.2 Natural Circulation Model

[(Proprietary information withheld under 10 CFR 2.390)

]

A.7.3 Approximated Analytical Solution of Flow Coast-Down

[(Proprietary information withheld under 10 CFR 2.390)

]

A.8 Secondary Steam System Model

[(Proprietary information withheld under 10 CFR 2.390)
]

A.8.1 Distribution of Steam Flow

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

A.8.2 Steam Relief and Safety Valves

[(Proprietary information withheld under 10 CFR 2.390)

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A.8.3 Steam Turbine Flow

[(Proprietary information withheld under 10 CFR 2.390)

]

A.9 Engineered Safeguards Features

[(Proprietary information withheld under 10 CFR 2.390)

]

A.10 Reactor Coolant System Small Break or Leak

[(Proprietary information withheld under 10 CFR 2.390)

]

A.11 Reactor Control System

[(Proprietary information withheld under 10 CFR 2.390)

]

A.12 Chemical Volume Control System

[(Proprietary information withheld under 10 CFR 2.390)

]

A.13 Reactor Protection System and ESFAS

[(Proprietary information withheld under 10 CFR 2.390)

]

A.14 Realistic Models - Descriptions and Model

[(Proprietary information withheld under 10 CFR 2.390)

]

A.15 Specific Computation Techniques

[(Proprietary information withheld under 10 CFR 2.390)

]

[(Proprietary information withheld under 10 CFR 2.390)

]

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6.0 CONCLUSIONS, LIMITATIONS AND CONDITIONS

The staff concludes that the Non-LOCA methodology as described in MUAP-07010-P (R2), "Non-LOCA Methodology," is acceptable for analysis of the Non-LOCA design basis events described in Chapter 15 of the US-APWR Tier 2 DCD with the following restriction:

- Since MARVEL-M has not been qualified for simulating the discharge of liquid through the pressurizer safety and relief valves, the usage of MARVEL-M is restricted to events which only discharge steam through the pressurizer safety and relief valves.

The following condition applies to the staff approval of the MUAP-07010:

- The Non-LOCA methodology described in MUAP-07010 Revision 4 is acceptable provided that it includes an approved method to model detailed core sub-channel thermal-hydraulic conditions. For the US-APWR DC application, that method is based on VIPRE-01M as documented in MUAP-07009, "Thermal Design Methodology," which is under NRC review.

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8.0 ACRONYMS

1D (or 1-D)	One-dimensional
3D (or 3-D)	Three-dimensional
AC	Alternating current
AFW	Auxiliary feedwater
ANS	American Nuclear Society
AOO	Anticipated operational occurrence
APWR	Advanced Pressurized Water Reactor
ASME	American Society of Mechanical Engineers
ARI	All rods in
ARO	All rods out
BOC	Beginning of cycle
CFR	Code of Federal Regulations
CHF	Critical heat flux
CLOF	Complete loss of flow
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
DCD	Design Control Document
DNB	Departure from nucleate boiling
DNBR	Departure from nucleate boiling ratio
DVI	Direct vessel injection
ECC	Emergency core cooling (or coolant)
ECCS	Emergency Core Cooling System
EFWS	Emergency Feedwater System
EOC	End of cycle
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
ESFAS	Engineered Safety Feature Actuation System
FCM	Fuel centerline melt
FWLB	Feedwater line break
GDC	General design criteria
HFP	Hot full power
HHIS	High-Head Injection System
HPI	High pressure injection
HPSI	High pressure safety injection
HZP	Hot zero power
IPP	Indian Point Plant
LBLOCA	Large break loss-of-coolant accident
LCO	Limiting condition for operation
LHS	Left hand side
LOCA	Loss-of-coolant accident
LOL	Loss of load
LWR	Light water reactor
MDC	Moderator density coefficient
MDNBR	Minimum departure from nucleate boiling ratio
MFW	Main feedwater
MHI	Mitsubishi Heavy Industries, Ltd.
MSDV	Main Steam Depressurization Valve

MSIV	Main Steam Isolation Valve
MSLB	Main steam line break
MSRV	Main Steam Relief Valve
NR	Neutron Reflector
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OECD	Organization for Economic Co-operation and Development
PA	Postulated accident
PAM	Post accident monitoring
PCMI	Pellet cladding mechanical interaction
PCT	Peak cladding temperature
PIRT	Phenomena identification and ranking table
PWR	Pressurized water reactor
QAP	Quality Assurance Program
RAI	Request for additional information
RCCA	Rod Cluster Control Assembly
RCP	Reactor coolant pump
RCS	Reactor coolant system
REA	Rod ejection accident
RG	Regulatory Guide
RHS	Right hand side
RIA	Reactivity initiated accident
RPS	Reactor Protection System
RPV	Reactor pressure vessel
RTD	Resistance temperature detector
RV	Reactor vessel
RWSP	Refueling Water Storage Pit
SAFDL	Specified acceptable fuel design limit
SBLOCA	Small break loss-of-coolant accident
SE	Safety evaluation
SER	Safety evaluation report
SG	Steam generator
SGTR	Steam generator tube rupture
SI	Safety injection
SIS	Safety Injection System
SLB	Steam line break
SRP	Standard Review Plan
TER	Technical evaluation report
TMI	Three Mile Island
TS	Technical specification
US-APWR	United States Advanced Pressurized Water Reactor
USNRC	United States Nuclear Regulatory Commission
WEC	Westinghouse Electric Corporation