15.0.0 Transient and Accident Analyses – Introduction (Related to NUREG-0800, Chapter 15, Section 15.0, "Introduction – Transient and Accident Analyses")

This chapter of the United States Advanced Pressurized-Water Reactor (US-APWR) Design Certification (DC) Safety Evaluation Report (SER) describes the United States Nuclear Regulatory Commission (NRC) staff evaluation of Mitsubishi Heavy Industries' (MHI) (the applicant's) analyses, presented in the US-APWR Design Control Document (DCD), of the plant's responses to postulated equipment failures or malfunctions. These analyses are used to determine the limiting conditions for operation, limiting safety system settings, and design specifications for safety-related components and systems.

15.0.0.1 Introduction

This section addresses DCD Tier 2, Sections 15.0.0.1, "Classification of Plant Conditions"; 15.0.0.2, "Plant Characteristics and Initial Conditions Assumed in the Accident Analyses"; 15.0.0.3, "Reactor Trip System and Engineered Safety Feature Systems Analytical Limit and Delay Times"; 15.0.0.4, "Component Failures"; 15.0.0.5, "Non Safety-Related Systems Assumed in the Analyses"; 15.0.0.6, "Operator Action"; 15.0.0.7, "Loss of Offsite AC Power"; 15.0.0.8, "Long Term Cooling"; and 15.0.0.9, "Pump Seal Cooling with Containment Isolation."

15.0.0.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided DCD Tier 2 descriptions in Sections 15.0.0.1 through 15.0.0.9, summarized as follows:

DCD Tier 2, Section 15.0.0.1, "Classification of Plant Conditions"

Each initiating event is categorized by frequency of occurrence as either an anticipated operational occurrence (AOO) or postulated accident (PA), as shown in Table 15.0-1, "Summary of Event Classification, Initial Conditions and Computer Codes." This classification provides the basis for the selection of applicable acceptance criteria.

DCD Tier 2, Section 15.0.0.2, "Plant Characteristics and Initial Conditions Assumed in the Accident Analyses"

Table 15.0-3, "Nominal Values of Plant Parameters," identifies the nominal values for plant parameters that are used for departure from nucleate boiling (DNB) events analyzed using the revised thermal design procedure (RTDP). For other events, the initial conditions are obtained by adding the maximum steady-state errors (for core power, average reactor coolant system (RCS) temperature and pressurizer pressure) to the rated values in the conservative direction. Unless otherwise stated, all reactor coolant pumps (RCPs) are assumed operational at the event initiation. All transients are assumed to begin with the most severe power distributions consistent with operation within the technical specifications (TS). The applicant states that each

event uses the bounding maximum or minimum value of the reactivity coefficients, even if the conservative combinations of parameters are not representative of realistic situations. The analyses assume the single highest-reactivity-worth rod cluster control assembly (RCCA) remains fully withdrawn during a reactor trip. Residual heat in a subcritical core is calculated in accordance with American Nuclear Society (ANS) standards.

DCD Tier 2, Section 15.0.0.3, “Reactor Trip System and Engineered Safety Feature Systems Analytical Limits and Delay Times”

Table 15.0-4, “Reactor Trip and ESF Actuation Analytical Limits and Time Delays Assumed for Transient Analyses,” summarizes the reactor trip and engineered safety feature (ESF) actuation analytical limits and response delay times for functions used in the event analyses. The DCD states that the analytical limits account for instrumentation channel and setpoint errors and the time delays are selected to give conservative results. Table 15.0-5, “Mitigation System Time Delays,” summarizes the time delays associated with accident-mitigating equipment.

DCD Tier 2, Section 15.0.0.4, “Component Failures”

Each event in the accident analyses incorporates the most limiting single active failure of a safety-related system as identified in Table 15.0-6 “Assumed Single Failures.” Operator errors are assumed as event initiators, but are not expressly accounted for in the analysis.

DCD Tier 2, Section 15.0.0.5, “Non Safety-Related Systems Assumed in the Analyses”

Only safety-related systems are credited in the US-APWR safety analyses. If nonsafety-related control systems will adversely impact the results, they are modeled in the evaluation with best estimate characteristics.

DCD Tier 2, Section 15.0.0.6, “Operator Action”

Operator actions are credited to mitigate the following accidents: inadvertent dilution of boron concentration in the RCS, steam generator (SG) tube failure, RCCA ejection, and failure of small lines carrying primary coolant outside containment. Operator action is also credited to prevent boric acid precipitation to assure long-term cooling after a LOCA.

DCD Tier 2, Section 15.0.0.7, “Loss of Offsite AC Power”

Both loss-of-offsite-power (LOOP) and offsite-power-available conditions are considered for each event that may be accompanied by a reactor or turbine trip. The US-APWR is designed such that the start of the RCP coast-down is delayed more than three seconds after the reactor/turbine trip. During this delay, the rods are inserted to the dashpot, which assures that the RCP flow reduction occurs after the limiting departure from nucleate boiling ratio (DNBR) has been reached. Because the minimum DNBR for the transient is the same with or without LOOP, the LOOP cases are generally not presented in the individual event evaluation sections.

DCD Tier 2, Section 15.0.0.8, “Long Term Cooling”

The reactor trip and ESF actuation systems are designed to mitigate accident conditions and to stabilize the plant at hot standby conditions. After the plant has been stabilized, the operators may transition to cold shutdown conditions using the residual heat removal system (RHRS).

Generally, the event specific discussions do not include this transition step, but they will include any assumptions regarding the actuation and operation of the RHRS.

DCD Tier 2, Section 15.0.0.9, "Pump Seal Cooling with Containment Isolation"

In the event of containment vessel (C/V) isolation, normal cooling to the RCP seal is lost. Because this could lead to seal degradation if the condition persists, the C/V isolation valves on the CCW supply and return lines are designed to be manually reopened from the main control room to restore RCP seal cooling.

Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC): The ITAAC associated with DCD Tier 2, Section 15.0.0, are given in DCD Tier 1, Section 2.4 (Reactor Systems), Section 2.5 (Instrumentation and Control) Section 2.7 (Plant Systems) and Section 2.9 (Human Factors Engineering).

Technical Specifications (TS): The following TS related to DCD Tier 2 Section 15.0.0 are listed in DCD Chapter 16:

- 2.1.1 Safety Limits
- 3.1.3 Moderator Temperature Coefficient
- 3.1.4 Rod Group Alignment Limits
- 3.1.5 Shutdown Bank Insertion Limits
- 3.1.6 Control Bank Insertion Limits
- 3.1.7 Rod Position Indication
- 3.2.1 Heat Flux Hot Channel Factor
- 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
- 3.2.3 Axial Flux Difference
- 3.3.1 Reactor Trip System Instrumentation
- 3.3.2 Engineered Safety Feature Actuation System Instrumentation
- 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
- 3.4.2 RCS Minimum Temperature for Criticality
- 3.4.4 RCS Loops – Modes 1 and 2
- 3.4.5 RCS Loops – Mode 3
- 3.4.6 RCS Loops - Mode 4
- 3.4.9 Pressurizer
- 3.4.10 Pressurizer Safety Valves
- 3.5.2 Safety Injection System - Operating
- 3.5.4 Refueling Water Storage Pit
- 3.7.1 Main Steam Safety Valves

3.7.2 Main Steam Isolation Valves

3.7.3 Main Feedwater Isolation Valves, Main Feedwater Regulation Valves, Main Feedwater Bypass Regulation Valves, and Steam Generator Water Filling Control Valves

3.7.4 Main Steam Depressurization Valves (MSDVs)

3.7.5 Emergency Feedwater System

Topical and Technical Reports:

MUAP-07008-P, "Mitsubishi Fuel System Design Criteria and Methodology," May 2007

MUAP-07009-P, "Thermal Design Methodology," May 2007

MUAP-07010-P, "Non-LOCA Methodology," July 2007

MUAP-07011-P, "Large Break LOCA Code Applicability Report for US-APWR," July 2007

MUAP-07013-P, "Small Break LOCA Methodology for US-APWR," July 2007

MUAP-07016-P, "US-APWR Fuel System Design Evaluation"

MUAP-07026-P, "Mitsubishi Reload Evaluation Methodology"

15.0.0.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR *[light-water reactor]* Edition," Section 15.0, "Introduction - Transient and Accident Analyses." Review interfaces with other Standard Review Plan (SRP) sections can also be found in NUREG-0800, Section 15.0.

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 10 CFR Part 50, Appendix A).
3. 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants."
4. 10 CFR Part 100, "Reactor Site Criteria."

The following General Design Criteria (GDC) from 10 CFR Part 50, Appendix A, are relevant to SRP Section 15.0:

1. GDC 2, as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.
2. GDC 4, as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and PA conditions, including such effects as pipe whip and jet impingement.
3. GDC 5, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.

4. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including AOOs.
5. GDC 13, as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.
6. 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.
7. 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. The safety function for each system (assuming the other system is not working) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the RCPB are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.
8. GDC 19, as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.
9. GDC 20, as it relates to the reactor protection system 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.
10. GDC 25, as it relates to the requirement that the reactor protection system 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.
11. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.
12. GDC 27 and 28, as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.
13. 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.
14. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.
15. GDC 34, as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.
16. GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.
17. GDC 55, as it relates to the isolation requirements for small-diameter lines connected to the primary system.
18. GDC 60, as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.

19. GDC 61, as it relates to the requirement that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and PA conditions.

Acceptance criteria adequate to meet the above requirements include:

For AOOs:

- Pressure in the reactor coolant and main steam 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 (B&PV) Code.
- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95 percent probability, with 95 percent confidence (95/95) DNBR limit.
- An AOO should not generate a PA without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

For PAs:

- Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.
- 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 these limits, then the fuel is assumed to have failed.
- The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.
- A PA 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 LOCAs:

- The calculated maximum fuel element cladding temperature shall not exceed 1,204 °C [2,200 °F].
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.0.0.4 Technical Evaluation

Classification of Plant Conditions
The staff finds that the applicant's categorization and classification of events in DCD Section 15.0.0.1 corresponds to the guidance in Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," and SRP Section 15.0. Meeting this guidance ensures that a broad spectrum of events has been considered and that the events are appropriately classified by frequency of occurrence as either AOO or PA.

Based on an issue initially raised during staff review of MUAP-07016-P, "US-APWR Fuel System Design Evaluation," the staff issued request for additional information (RAI) 297-2287, Question 15.0.0-13, regarding the potential need for evaluation of RCP over-speed as an AOO initiating event. In a July 3, 2009, response, the applicant stated that (1) evaluation of RCP over-speed events is not required by the SRP and (2) the US-APWR is no different in this respect from all PWRs that use synchronous RCP motors. The applicant response discussed how RCP over-speed events are bounded by other analyzed events from the perspective of RCS cooldown and pressurization effects. The applicant response also provided comparisons of effects of cold versus hot conditions relative to fuel assembly lift-off considerations. The staff agrees with this assessment and is satisfied that RCP over-speed does not need to be considered an AOO initiating event.

The staff reviewed the acceptance criteria for AOOs, PAs, and LOCAs and found they were consistent with SRP Section 15.0. The staff issued RAI 297-2287, Question 15.0.0-16, requesting the applicant provide numerical values for the minimum DNBR and peak primary and secondary coolant system pressures for each event. The response, dated December 20, 2011, included Table 15.0.0-16.1, "Results of Chapter 15 Accident Analyses Compared to Acceptance Criteria," comparing the analysis results to the AOO acceptance criteria values (which are more limiting than the PA acceptance criteria values). The acceptance criteria for the 95/95 DNBR limit was identified as 1.45, which is consistent with DCD Section 4.4.1.1.2, and more limiting than the TS 2.1.1.1 limiting condition for operation (LCO). The 110-percent design values for the peak primary and secondary pressures were identified as 18.96 megapascals (MPa) [2750 pounds per square inch absolute (psia)] and 9.10 MPa [1320 psia], consistent with DCD Sections 5.3.3 and 10.3.2. The response demonstrated that the analyses meet all three AOO criteria except for the minimum DNBR resulting from RCP rotor seizure (Section 15.3.3), single RCCA withdrawal (Section 15.4.3) and RCCA ejection (Section 15.4.8) events. This is acceptable because these events are PAs, which allow a percentage of fuel to fail. The event specific analyses of these accidents describe how all PA criteria, including radiological consequences, are met.

<u>Plant Characteristics and Initial Conditions</u>
Assumptions regarding the plant characteristics and initial conditions used in the analyses are provided in DCD Section 15.0.0.2 and Tables 15.0-1 through 15.0-3. The staff did not feel this information was comprehensive and issued RAI 297-2287, Question 15.0.0-10, requesting more details on the initial conditions and key input parameters assumed for each event. The applicant's response, dated December 20, 2011, provided this information on an event-specific basis in Tables 15.0.0-10.1, "Summary of Key Input Parameters," and 15.0.0-10.2, "Summary of Key Input Parameters." The response confirmed that DNB-limited events analyzed using the RTDP are initiated from nominal conditions. The staff finds this appropriate because the RTDP is designed to address uncertainties with a statistical combination technique. For other events (non-DNB-limited analyses or DNB calculations that do not use the RTDP), the DCD states that the maximum steady-state errors for core power, average RCS temperature, pressurizer pressure, and RCP flow are added in the conservative direction.

The staff confirmed this statement was true for non-DNB-limited analyses and issued RAI 769-5797, Question 15.0.0-28, to determine if uncertainties were included for DNB not analyzed with the RTDP. On July 15, 2011, the applicant responded that the only events that fall in this category are the inadvertent opening of a SG relief or safety valve (Section 15.1.4) and Cases A and B of the main steam line break (MSLB) (Section 15.1.5). While these events were run at nominal conditions for the DCD analysis, the response included sensitivity studies demonstrating that the impact of the steady-state errors is small relative to the margin to the acceptance criteria. The staff finds this response acceptable because the limiting cooldown event (Case C of the MSLB in Section 15.1.5) bounds the results of the sensitivity study.

As stated in the applicant's response to RAI 297-2287, Question 15.0.0-10, all Chapter 15 transients assume 10 percent of the SG tubes are plugged. This is non-conservative for the increase in heat removal transients analyzed in DCD Sections 15.1.1 through 15.1.5 because it reduces the heat transfer from the primary to secondary system, lessening the severity of these events. The staff requested this assumption be justified in RAI 769-5797, Question 15.0.0-29. The applicant's September 9, 2011, response concluded that the impact of the SG tube plugging is negligible, based on sensitivity studies of the DNBR limiting cooldown event, the Case C MSLB (Section 15.1.5). Due to the relatively small margin to the DNBR limit in this analysis, the staff was unable to reach the same conclusion; therefore, a follow up RAI was issued and resolved as described in Section 15.1.5.4 of this SER. The staff was satisfied that the response to the original RAI demonstrated SG tube plugging is not a key parameter for the Section 15.1.1-15.1.4 events because all of these have significantly more margin to the DNBR limit.

The staff reviewed the steady-state errors provided by the applicant in DCD Section 15.0.0.2.2 and found the ± 2-percent allowance for core power to be acceptable because the magnitude is consistent with SRP Section 15.1.1-15.1.5 guidance for analytical models and because the tolerance is always added in the positive direction (the minimum power level for events initiated from power is 100 percent core power). Because the DCD did not provide a basis for the assumed steady-state variations in pressure and temperature, it was requested in RAI 769-5797, Question 15.0.0-27. In a response dated September 9, 2011, the applicant explained that the steady-state variations are determined by adding margin to the uncertainties calculated using a proprietary method similar to that described in MUAP-09022-P, "US-APWR Instrument Setpoint Methodology." The staff confirmed that the detailed calculations are consistent with MUAP-09022-P methodology and that the results bound the errors used in the DCD. Based on this, the staff finds that the steady-state errors for pressure and temperature are suitably conservative.

The staff asked for justification that the steady-state errors were added in the conservative direction in RAI 297-2287, Question 15.0.0-8. In a July 3, 2009, response, the applicant summarized sensitivity studies run with uncertainties on power, RCS temperature and RCS pressure added in the direction opposite what was used in the DCD. For all but two events, the DCD was demonstrated to be bounding for each event-specific acceptance criterion. The response also explained why the two exceptions were justified. The staff reviewed the proprietary discussion and agrees that appropriate conservatisms were used in the selection of steady-state errors for all Chapter 15 transients.

In RAI 297-2287, Question 15.0.0-6, the staff requested that the applicant confirm the event scenarios assumed in the DCD bound all operating conditions with respect to mode and power level.

In a July 3, 2009, response, the applicant provided Table 15.0.0-6.1, "DCD Limiting Case Selection Matrix," which explains why the plant operating mode assumed in the DCD analysis is the most limiting for each event. The staff's evaluation of this response is included in the event specific discussions of this SER.

In RAI 297-2287, Question 15.0.0-7, the staff requested the applicant confirm that the plant characteristics assumed in the transient analysis are consistent with the range of values specified in the TS. In a July 3, 2009 response, the applicant stated that all but one of the TS related to the Chapter 15 analysis were considered in developing the analysis assumptions. The exception was pressurizer water level, where the transients were initiated at the nominal operating level plus uncertainty (47.6 percent) rather than the maximum level specified in TS 3.4.9 (92 percent). The staff issued RAI 399-2992, Question 16-298 to express concern that a heatup AOO initiated at the maximum TS LCO could lead to a water solid pressurizer with liquid or two-phase release through the pressurizer safety valves. Unless the safety valves and associated piping are qualified for liquid or two-phase flow, this scenario could propagate to a more severe event, which would violate the acceptance criteria for AOOs. In a response dated October 6, 2011, the applicant proposed reducing the TS LCO for Mode 1 to 60 percent. The applicant then identified the two limiting pressurizer overfill AOO events (loss of alternating current (ac) power in DCD Section 15.2.6 and loss of normal feedwater flow in DCD Section 15.2.7) and initiated these analyses with a pressurizer water level of 60 percent to demonstrate there is sufficient margin between the maximum pressurizer water level and the location of the safety valves. The response proposed DCD changes to incorporate the new TS LCO and analyses. The staff agrees that the limiting AOO events were correctly identified and the analyses confirmed no liquid or two-phase flow will be relieved through the pressurizer safety valves for an event initiated at the TS LCO.

The applicant also proposed to revise the CVCS malfunction event (Section 15.5.2) to terminate on automatic isolation of the CVCS on a high pressurizer water level signal rather than on operator action. The proposed DCD changes involve Tier 1, Table 2.5.4.3 and Tier 2, Chapters 7 and 15. The staff finds this acceptable, as discussed in Section 15.5.1.4 of this SER.

The applicant acknowledged that while water relief through the pressurizer safety valves does not violate any acceptance criteria for PAs, there is a US-APWR requirement in DCD Section 5.4.10.1 which states that the pressurizer is designed to prevent water relief through the safety valves following a feedwater line rupture. The applicant stated that the feedwater line rupture analysis (Section 15.2.8) demonstrates this requirement is met when the accident is initialized at the nominal operating level plus uncertainty. However, if the water level is assumed to be at the

proposed TS Mode 1 LCO of 60 percent, the pressurizer may fill and relieve water through the safety valves. Because this water relief would happen at a time well after the reactor trip (when the core power is at decay heat levels) the amount of water relieved will be bounded by the LOCA analysis presented in Section 15.6.5. The applicant proposed DCD changes to the design requirement (Section 5.4.10.1) and the feedwater break analysis (Section 15.2.8) to clarify that the feedwater break is to be initiated from an initial pressurizer water level that is less than or equal to the nominal level plus instrument uncertainty. The staff agrees that water relief is allowed for PAs and finds the proposed changes regarding the feedwater break analysis to be acceptable. The staff will use **Confirmatory Item 15.00-1** to track that all changes proposed in the response to RAI 399-2992, Question 16-298 are included in the next DCD revision.

In response to the original request for confirmation that the plant characteristics assumed in the transient analysis are consistent with the range of values specified in the TS (RAI 297-2287, Question 15.0.0-7), the applicant stated that all six main steam safety valves (MSSV) are modeled as a single valve set to open at 8.52 MPa [1236 psia]. This value does not bound the three MSSV lift settings defined in TS 3.7.1 (8.150,8.377,8.577 MPa [1182,1215,1244 psig] plus 1-percent uncertainty); therefore, the staff asked for justification in RAI 809-5957, Question 15-34. In a September 30, 2011, response, the applicant presented a sensitivity analysis for the loss-of-external-load (LOEL) event (DCD Section 15.2.1) using an MSSV model based on the three setpoints with uncertainty as defined in the TS. Compared to the simplified model, the DNBR and peak RCS pressure were relatively unchanged, but the peak secondary-side pressure was higher. This finding led to the applicant incorporating the detailed three-setpoint MSSV model (with uncertainties) into the two limiting secondary-side pressure events (LOEL in DCD Section 15.2.1 and turbine trip in DCD Section 15.2.2). The proposed changes to the DCD are not associated with this RAI, but are included in response to RAI 809-5957, Question 15-33 (discussed below), and RAI 789-5920, Question 15.02.01-15.02.05-9 (discussed in Section 15.2.1.4 of this SER).

In RAI 809-5957, Question 15-33, the staff asked if any of the events were designed to maximize peak secondary-side pressure. In a September 30, 2011 response, the applicant stated the limiting event for secondary-side pressure is the LOEL event (DCD Section 15.2.1). Based on sensitivity studies, the applicant concluded that the DCD assumptions on the MSSV settings and RCS temperature are not conservative with respect to secondary-side pressure. As such, the applicant proposed adding a new case to DCD Section 15.2.1, specifically designed to maximize secondary-side pressure. This case will be similar to the existing RCS pressure case except that it will use a detailed, three-setpoint MSSV model and a higher initial RCS temperature. The staff reviewed the analysis and agrees that the proposed DCD case accurately captures the maximum secondary-side pressure. Inclusion of the changes to Chapter 15 in the next DCD revision will be tracked as **Confirmatory Item 15.00-2**.

For non-LOCA analysis, the applicant states that Doppler reactivity includes a Doppler power coefficient of reactivity and a Doppler fuel temperature coefficient of reactivity. Because the DCD only addresses the Doppler power coefficient of reactivity, the staff issued RAI 786-5881, Question 15.0.0-30, requesting information on what values were used for the Doppler fuel temperature coefficient of reactivity for each event. In an August 24, 2011 response, the applicant explained that the Doppler fuel temperature coefficient is generally chosen to correspond to the Doppler power coefficient; either both are minimum or both are maximum. The staff agrees with this approach because the applicant's definition of Doppler reactivity implies the coefficients will be at the same extreme. The only exceptions are the overcooling events from hot zero power (inadvertent opening of a SG relief or safety valve in Section 15.1.4 and Cases A and B of the MSLB in Section 15.1.5). In these analyses, the maximum negative

Doppler temperature coefficient is combined with the minimum Doppler power coefficient. This is conservative because the former will add more reactivity to the core as coolant temperature decreases while the latter will subtract less reactivity from the core as power increases. The applicant proposed adding this explanation to the DCD, along with the actual values used for the Doppler fuel temperature coefficient of reactivity. The staff agrees with the proposed changes and will use **Confirmatory Item 15.00-3** to track their inclusion in the next DCD revision.

RAI 786-5881, Question 15.0.0-30, also asked how the values selected for the Doppler reactivity coefficients were determined to be bounding for the DCD Section 15.2 heatup events and the DCD Section 15.3 low-RCS-flow events. The response stated that maximum Doppler feedback was assumed for all 15.2 and 15.3 events except that the LOEL event (DCD Section 15.2.1) assumed minimum Doppler feedback. Based on sensitivity studies for each event, the applicant concluded that the Doppler feedback is not a key parameter for any of these events.

The staff agrees with this statement for the LOEL (DCD Section 15.2.1), loss-of-ac-power (DCD Section 15.2.6), loss-of-feedwater-flow (DCD Section 15.2.7), and loss-of-reactor-coolant-flow (DCD Section 15.3.1) events. However, because the results of the sensitivity studies did not include how DNBR was affected for the limiting AOO (DCD Section 15.2.2) or limiting PA (DCD Section 15.3.3), the staff requested this be provided in RAI 864-6150, Question 15-35. In a December 7, 2011, response, the applicant provided additional information. For the DCD Section 15.2.2 event, the DCD assumption of minimum feedback was found to be bounding for DNBR. For the 15.3.3 event, the minimum feedback assumed in the sensitivity study was found to predict slightly more fuel rods with DNB failure than the maximum feedback assumed in the DCD, but the results were less than the 10-percent failed fuel rods assumed in the radiological evaluation. The applicant noted that the DNBR analysis was conservatively run with constant RCS pressure. Another sensitivity study was performed using the transient RCS pressure, and these results predicted fewer failed fuel rods in DNB failure than in the DCD case. The staff agrees this demonstrates the combination of parameters assumed in the DCD is suitably conservative.

The DCD states a conservative bottom-skewed axial power distribution is used to help define the control rod insertion worth. The staff issued RAI 297-2287, Question 15.0.0-17, requesting that the applicant describe how this distribution is determined. The applicant responded that the axial power distribution was selected to bound that allowed by the TS during normal operational conditions. The staff agrees a bottom-skewed axial power distribution is conservative because it delays the insertion of negative reactivity and the selection of a distribution that bounds those allowed by TS is appropriate.

<u>Set Points</u>
The reactor trip system (RTS) and ESF system activation limits and delay times assumed in the analyses are provided in DCD Section 15.0.0.3. The staff confirmed that the RTS and ESF analytical set points and response time delays identified in DCD Table 15.0-4 are consistent with the information presented in DCD Table 7.2-3, “Reactor Trip Variables, Ranges, Accuracies, Response Times, and Setpoints (Nominal),” Table 7.3-4, “Engineered Safety Features Actuation Variables, Ranges, Accuracies, Response Times, and Setpoints (Nominal),” and in TS 3.3.1 and 3.3.2. The staff did not understand the basis or starting point for the time delays identified for the mitigating systems in DCD Table 15.0-5, and issued RAI 769-5797, Question 15.0.0-26, requesting this information. The applicant’s September 9, 2011, response stated that the time delays listed in the first four rows of DCD Table 15.0-5 are mechanical valve closure times that are to be added to the signal delays in DCD Table 15.0-4 to determine the

total delay time. The closure times for the main feedwater isolation valve, main feedwater regulation valve and main steam isolation valve (MSIV) are consistent with surveillance requirements (SR) in TS 3.7.3 and TS 3.7.2 and the closure times for the main steam relief valve block valve closure and emergency feedwater isolation valve are based on experience and procurement specifications. The applicant explained the last three rows of DCD Table 15.0-5 are mitigating systems involving pump startups and the total delay time consists of several components. A breakdown of these components was provided, and for each case, the signal delay from Table 15.0-4 was included in calculation of the overall delay. The applicant proposed changes to the DCD to explain which of the delays from Table 15.0-5 include the signal delays from Table 15.0-4 and which do not. The staff finds this portion of the response acceptable because the applicant provided a basis and starting point for the time delays identified in Table 15.0-5, and proposed changes to the DCD to explain the starting points.

The response to RAI 769-5797, Question 15.0.0-26, also clarified that the main steam line pressure actuation signal for ECCS pump start up and MSIV closure includes lead/lag compensation. As such, it will occur earlier in the transient than the main steam line pressure actuation signal for emergency feedwater (EFW) isolation, which does not have lead/lag compensation. The applicant proposed changes to DCD Sections 15.1.4, 15.1.5, and 15.2.8 to clarify when the lead/lag compensation was used on the main steam line pressure signal. The staff confirmed the response was consistent with the functional logic diagrams in Figure 7.2-2, "Functional Logic Diagram for Reactor Protection and Control System" of the DCD and found the proposed changes acceptable, except those in Section 15.1.4.

In the 15.1.4 event, ECCS is actuated on low pressurizer pressure, but the proposed changes appear to state that ECCS is actuated on a main steam line pressure signal with lead/lag compensation. The staff issued RAI 864-6150, Question 15-36, asking for an explanation. In a response dated December 7, 2011, the applicant explained that low main steam line pressure is available to actuate ECCS, even though it is not the first to occur. Because it is available, the applicant thought it was relevant to describe how the signal was compensated. The staff is satisfied with this explanation. Inclusion of the proposed changes from the original query, RAI 769-5797, Question 15.0.0-26, will be tracked as **Confirmatory Item 15.00-4.**

The staff noted that DCD Tables 15.0-4 and 15.0-5 do not include actuation analytical limits and time delays for main feedwater isolation on a high-high SG water level signal (credited in the increase in feedwater flow in DCD Section 15.1.2) or CVCS isolation on a high pressurizer water level (credited in CVCS malfunction that increases RCS inventory in Section 15.5.2). The staff issued RAI 882-6237, Question 15-38, requesting this information, and the response will be tracked as **Open Item 15.00-1**.

<u>Limiting Single Failure</u>
Information on component failures assumed in the safety analyses is provided in DCD Section 15.0.0.4. The staff issued RAI 297-2287, Question 15.0.0-11, requesting an explanation for how the assumed single failures identified for each event in Table 15.0-6 were determined to be limiting. In a response dated July 3, 2009, the applicant stated that the first step was to identify the mitigating systems assumed in the safety analysis for each event, and these were provided in Table 15.0.0-11.1, "Mitigative Systems Assumed in the Chapter 15 Safety Analysis." The second step was to determine the single failure assumption for each mitigating system, and these were provided in Table 15.0.0-11.2, "Potential Effect of Single Failure Assumption." The final step was to perform an event-specific comparison to find the single failure that results in the most severe analysis outcome. Because CVCS isolation was added as a mitigating system for a CVCS malfunction event that increases RCS inventory

(Section 15.5.2) subsequent to the RAI response, it was not included in the original analysis. The staff issued RAI 882-6237, Question 15-39, requesting a single-failure analysis of CVCS isolation and will track the response as **Open Item 15.00-2**. Staff evaluation of the original RAI response is in the event-specific discussions included in this SER.

The staff issued RAI 297-2287, Question 15.0.0-14, to address single failure of the emergency feedwater system (EFWS) pumps. DCD Table 8.3.1-4, "Electrical Load Distribution - Class 1E GTG Loading," shows four 50 percent divisions of electrical safety equipment. The two motor-driven emergency feedwater pumps (MDEFWPs) are powered by Division B and C Class 1E power supplies, respectively. If it is assumed that one division of electrical power supply is out for maintenance, as allowed by TS, and that a single failure is experienced on the other division, both MDEFWPs would be inoperable during design-basis events. The applicant was asked to discuss the operability and adequacy of the two turbine-driven emergency feedwater pumps (TDEFWPs) with respect to the availability of steam supplies and the arrangement of the feedwater flow system. In response, the applicant provided details regarding the design features of the EFWS, which assure that the TDEFWPs will perform the required safety-related decay heat removal function in case both of the MDEFWPs become inoperable. The staff found this response acceptable because it provided the requested information.

<u>Non-Safety-Related Systems Assumed in the Analysis</u>
Information on nonsafety-related systems assumed to be active in the analyses is provided in DCD Section 15.0.0.5. The application states that nonsafety-related systems are not required to mitigate the consequences of events. This satisfies the requirement that only safety-related systems or components be used in analyses evaluating the mitigation of AOOs and PAs. The application also states that nominal control system characteristics are modeled in the accident analyses only if they adversely impact the results. Staff evaluation of this statement is in the event-specific discussions included in this SER.

<u>Operator Actions</u>
Information on operator actions assumed in the analyses is provided in DCD Section 15.0.0.6. Operator actions are credited for inadvertent decrease in boron concentration (DCD Section 15.4.6), radiological consequences of SG tube failure (DCD Section 15.6.3), control rod ejection accidents (DCD Section 15.4.8) and failure of small lines carrying primary coolant outside the containment vessel (DCD Section 15.6.2). While operator action was originally credited for a CVCS malfunction that increases RCS inventory (DCD Section 15.5.2), this is no longer necessary (as discussed in Section 15.5.2.4 of this SER) and will be removed from the DCD in Confirmatory Item 15.00-1. The staff confirmed that the operator actions credited in the event-specific analysis are consistent with DCD Table 7.5-5, "List of Accidents and Credited Manual Actions," which also identifies the associated alarm. DCD Section 7.5.1.5 states that all credited manual operator actions are included in the human factor engineering (HFE) program described in Chapter 18.

Because the staff could not determine how future plant emergency operating procedures (EOPs) would be verified against the operator actions credited in Chapter 15, this information was requested in RAI 297-2287, Question 15.0.0-12. In response, the applicant stated that the plant-specific EOPs will be based on an Emergency Response Guidelines (ERG) document. The ERG are being developed by MHI and will address all operator actions assumed in the safety analysis. The applicant stated that the responsibilities of the combined license (COL) applicant to develop and implement EOPs in COL Item 13.5(6) are adequate to demonstrate

that the EOPs will be consistent with the credited operator action. Additionally, the staff notes that DCD Tier 1, ITAAC Table 2.9-1, "Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria," Item 10, requires the verification and validation (V&V) program to be conducted in accordance with the V&V implementation plan. Part of the V&V program will be operator action completion times assumed in the safety analysis, verified through integrated system validation as described in DCD Section 18.10.2.3. The staff is satisfied that COL Information Item 13.5(6) is adequate to direct the COL applicant to demonstrate that operator actions and completion times are consistent with those assumed in the design-basis analysis.

<u>Loss of Offsite Power</u>
Information related to the assumed availability of offsite power in the analyses is provided in DCD Section 15.0.0.7. In addition to the limiting single failure assumed in the event, the applicant considers LOOP to be a secondary effect resulting from grid disturbances caused by a turbine-generator trip. In the event of a LOOP, the electrical system is designed such that power to the RCPs can be maintained for at least 3 seconds after the reactor/turbine trip.

In DCD Chapter 15, the applicant attributed the delay to the time it would take for a grid instability to propagate to the plant offsite power source, but in DCD Chapter 8, the applicant attributed the delay to the large inertia of the turbine-generator. The staff issued RAI 687-5394, Question 15.0.0-24, asking for clarification. In a response dated September 9, 2011, the applicant proposed adding the detailed basis for the assumed 3-second delay to DCD Section 8.2.3 and revising Chapter 15 to reference Section 8.2.3. The staff agrees that Chapter 8 is a more appropriate place to describe operation of the electrical distribution system. Section 8.2 of this SER includes an evaluation of the 3-second delay bases. The staff will track inclusion of the proposed changes to the next DCD revision as **Confirmatory Item 15.00-5.**

The DCD states that the 3-second delay between the reactor/turbine trip and LOOP allows enough time for the rods to be inserted to the dashpot and stop the excursion before the RCPs begin coast-down, decoupling the minimum DNBR from the availability of offsite power. The staff issued RAI 297-2287, Question 15.0.0-3, requesting a demonstration that the minimum DNBR is independent of a LOOP. In a July 3, 2009, response, the applicant provided a LOOP sensitivity analysis. Four representative DNBR events (rod withdrawal at power for 75- and 5.0-pcm (percent millirho)/second withdrawal rates in DCD Section 15.4.2, Case C of the MSLB in DCD Section 15.1.5, and LOEL in DCD Section 15.2.1) were evaluated and the results confirmed that the minimum DNBR for the transient is the same when there is no LOOP as when there is a 3-second delay to LOOP. Additional evidence that a LOOP with a 3-second delay has the same minimum DNBR as the no-LOOP case is found in the response to RAI 301-2324, Question 15.1-1, for an increase-in-feedwater-flow event (Section 15.1.2) and in the response to RAI 306-2333, Question 15.3.1-1, for a partial loss-of-forced-reactor-coolant-flow event (DCD Section 15.3.1.1).

The DCD states that the time delay between the reactor/turbine trip and LOOP is not a key parameter for peak pressure analyses, and either 0 or 3 seconds is assumed, as described in the applicable subsection. The sensitivity studies included in the response to RAI 297-2287, Question 15.0.0-3, demonstrate that the peak pressure is the same with no LOOP as with a 3-second delay to LOOP. The response also confirms it is conservative to assume a 0-second delay to LOOP (as done for the feedwater line break analysis in DCD Section 15.2.8) because this results in a slightly higher peak pressure than the no-LOOP (or 3-second delay to LOOP). Plots of sensitivity studies from the RAI 297-2287, Question 15.0.0-3, response regarding the limiting event for peak secondary-side pressure (LOEL in DCD Section 15.2.1) demonstrate that

LOOP has no impact on the peak secondary-side pressure. The transient pressure curves are indistinguishable between cases of no-LOOP and LOOP with 0- or 3-second delays.

The response to RAI 297-2287, Question 15.0.0-3, also demonstrates the no-LOOP case bounds the LOOP with a 3-second delay for fuel centerline temperature, peak fuel enthalpy and peak cladding temperature. The fuel centerline temperature study was based on the limiting rod ejection event, hot zero-power, end-of-cycle (HZP-EOC) (DCD Section 15.4.8). The sensitivity study for peak fuel enthalpy included the two most limiting events, rod ejections at HZP–EOC and HZP-beginning-of-cycle (BOC) (DCD Section 15.4.8) and the peak cladding temperature study was based on RCP rotor seizure (DCD Section 15.3.3).

The staff finds the response to RAI 297-2287, Question 15.0.0-3, acceptable because the limiting events were correctly identified and the supporting analyses demonstrated the no-LOOP case bounds LOOP with a 3-second delay for DNBR, peak RCS and secondary-side pressures, fuel centerline temperature, peak fuel enthalpy and peak cladding temperature.

<u>Long-Term Cooling</u>
Information on long-term cooling to stabilize the plant by automatic systems and operator actions is provided in DCD Section 15.0.0.8. The reactor trip and engineered safety features are designed to mitigate accidents and stabilize the plant at hot-standby conditions. Afterward, the operators may maintain hot-standby conditions, during which core decay heat is removed through the SG, or transition to cold-shutdown conditions, during which core decay heat is removed by the RHRS. Functions performed by the RHRS include cooling of the RCS and cooling of the containment. DCD Section 15.0.0.8 defines safe shutdown for US-APWR as achieving cold-shutdown conditions following design-basis events and AOOs using safety-related systems.

The Chapter 15 safety analysis evaluations are generally analyzed only long enough to assure that the acceptance criteria primarily challenged by the specific events have been met. The analyses therefore do not typically address the transition to the safe shutdown using the RHRS, unless event-specific analysis assumptions are made regarding RHRS actuation and operation.

The principal exception is for post-LOCA, long-term cooling, which is extensively addressed in DCD Section 15.6.5. During LOCAs, both the RCS cooling and containment cooling functions of the RHRS are employed. Staff evaluation of post-LOCA, long-term cooling is provided in Section 15.6.5.4.3 of this SER.

<u>Pump Seal Cooling with Containment Isolation</u>
A description of RCP shaft seal cooling following isolation of the containment is included in DCD Section 15.0.0.9 and referenced to DCD Section 9.2.2. Staff evaluation of this system is included in Section 9.2.2 of this SER.

<u>Review of MHI Reload Evaluation Methodology</u>
DCD Chapter 15 contains the reference safety analysis for the US-APWR initial plant configuration and core. Any changes to the initial design must be evaluated to verify that the reference safety analysis limits are met, or to identify which transients need to be reevaluated. This practice is referred to as a reload evaluation and the process proposed for the US-APWR is documented in Technical Report MUAP-07026-P "Mitsubishi Reload Evaluation Methodology." While this document is identified in TS 5.6.3, the staff thought it should be evaluated as part of Chapter 15, and issued RAI 786-5881, Question 15.0.0-31, requesting that MUAP-07026-P be referenced in Chapter 15. In a response dated August 24, 2011, the

applicant proposed DCD changes to add this document as DCD Reference 15.0-21, and describe it in DCD Section 15.0.0.2. The staff is satisfied with this response and will track the proposed changes to DCD Chapter 15 as **Confirmatory Item 15.00-6.**

The staff generated RAI 882-6237, Question 15-37, asking how changes made to the DCD since MUAP-07026-P was initially issued in 2007 will be incorporated. Subsequent to this RAI, the staff learned that a revised document was issued in August 2011. As such, **Open Item 15.00-3** will be used to track staff evaluation of Revision 1 and the response to RAI 882-6237, Question 15-37.

The MHI methodology for performing the safety evaluation of reload cores uses the reference analysis (as documented in the plant-specific final safety analysis report (FSAR), derived from the US-APWR DCD Chapter 15 safety analysis) and a set of key safety parameters (defined for each transient in MUAP-07026-P, Section 3.3). In the reference analysis, the values of the key safety parameters are selected to encompass those expected in subsequent cycles. For each reload, the key safety parameters for each event are reevaluated and if the key safety parameters are bounded, the reference safety analysis remains valid and no further work is needed.

If a reload safety parameter is not bounded, a reevaluation is performed which is either a re-analysis of the transient or a quantitative evaluation that conservatively evaluates the magnitude of the effect and explains why the actual analysis of the events do not need to be repeated. This approach is consistent with the “bounding analysis” concept that was approved for a different vendor in Westinghouse Commercial Atomic Power (WCAP)-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology.” The staff issued RAI 297-2287, Question 15.0.0-18, requesting clarification of the extent of future design changes to which the reload methodology would remain applicable. In response, the applicant stated that it would not be necessary to change the reload methodology as long as the safety analysis methodology and analysis codes are not changed. The staff finds this acceptable because the analysis codes influence the choice of key safety parameters, so they are considered part of the reference methodology.

The staff reviewed the safety evaluation phase of the reload design process (Section 2.4 of MUAP-07026-P) and confirmed that the acceptance criteria that must be demonstrated in order for a design to be “final” are consistent with SRP Section 15.0. The staff reviewed the accident analysis methods described in MUAP-07026-P, Section 3.2.1 through 3.2.8, and confirmed they were consistent with the DCD. The staff confirmed that the computer codes identified in MUAP-07026-P, Section 3.2.9, are the same codes identified in the DCD. NRC approval of these codes is discussed in Section 15.0.2.4 of this SER. The staff issued RAI 297-2287, Question 15.0.0-20, requesting that the applicant clarify the description of how the TWINKLE-M computer code (See description under Section 15.0.2.2 of this SER) accounts for Doppler and moderator feedback effects. In a response dated July 3, 2009, the applicant explained the methodology and acknowledged that the description in MUAP-07026-P, Section 3.2.9.2, was incomplete. The applicant proposed corrective changes to MUAP-07026-P, but these were not incorporated into Revision 1 of the document. The staff will track incorporation of these changes into the next revision of MUAP-07026-P as part of **Open Item 15.00-3**.

The staff issued RAI 297-2287, Question 15.0.0-21, requesting clarification of the basis and the value used for the conservative multiplier applied to the total rod worth described in MUAP-07026-P, Section 4.2.2.1. The staff question was prompted by an apparent discrepancy between the discussion provided in this section and DCD Section 15.0.0.2.5. The July 3, 2009, applicant response described the technical basis for selecting the conservative multiplier,

resolved the apparent discrepancy with the DCD wording, and provided the value of the conservative multiplier used in the analyses.

The staff issued RAI 297-2287, Question 15.0.0-23, requesting clarification for the discussions regarding axial power distributions and fuel temperatures in MUAP-07026, Sections 5.3.1.2 and 5.3.2. These discussions indicate that if a parameter is bounded by the reference case, then the normal operation and AOO analysis previously performed remain acceptable. However, a question arose regarding whether this applies to PAs as well. In response, the applicant stated that the axial power distributions and fuel temperatures discussed in Sections 5.3.1.2 and 5.3.2 of MUAP-07026-P are meant also to apply to PAs, and that MUAP-07026-P will be revised to discuss PAs in general as well as event-specific axial power distributions for certain PAs, but these changes were not incorporated into Revision 1 of the document. The staff will track incorporation of a PA discussion in the next revision of MUAP-07026-P as part of **Open Item 15.00-3**.

15.0.0.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19" that affect this section.

15.0.0.6 Conclusions

As a result of the open and confirmatory items, the staff is unable to finalize its conclusion on Section 15.0 in accordance with the requirements of NRC regulations, including but not limited to 10 CFR Part 20, 10 CFR Part 50 (especially 10 CFR 50.46 and 10 CFR Part 50, Appendix A), 10 CFR Part 100 and 10 CFR Part 52.

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms

Although the US-APWR design utilizes the alternative source term (AST) methodology, SRP Section 15.0.1 is focused on the application of AST to operating reactors and is not applicable to the US-APWR. See Section 15.0.3 of this SER for the details of the radiological consequence analyses for the US-APWR.

15.0.2 Review of Transient and Accident Analysis Methods

15.0.2.1 Introduction

This section discusses the safety analyses methodology.

15.0.2.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: Tier 2, Section 15.0.2.1, identifies the MHI topical reports that are relevant to the safety analysis and DCD Tier 2, Section 15.0.2.2, describes the following principal computer codes used in the accident analyses:

<u>MARVEL-M</u> is a multi-loop plant system transient analysis computer code used to calculate the transient behavior of the pressurized-water reactor system. The program models the reactor core, reactor vessel, each of the four reactor coolant loops, the four SGs and associated systems. It simulates the reactor kinetics and the thermal-hydraulics of the RCS, the pressurizer, the secondary steam and feedwater systems, the reactor control and protection system, and selected engineered safeguards systems.

<u>VIPRE-01M</u> is a subchannel, thermal-hydraulic analysis code with steady-state and transient capabilities. It calculates time-dependent changes in minimum DNBR and other parameters.

<u>TWINKLE-M</u> is a multidimensional, spatial neutron kinetics code that solves the two-group transient diffusion equations using a finite difference technique. This code is used to analyze changes in dynamic behavior of space- and time-dependent neutron flux in response to reactivity accidents.

<u>RADTRAD</u> is a computer model for estimating doses at offsite locations such as the exclusion area boundary (EAB) and the low-population zone (LPZ), as well as onsite locations due to postulated radioactivity releases from design basis accident conditions.

<u>ANC</u> is a three-dimensional two-group diffusion core calculation code based on a nodal expansion method. It calculates nuclear parameters, such as local peaking factors.

<u>WCOBRA/TRAC (M1.0)</u> is used for the calculation of thermal-hydraulic behavior during a large break LOCA. The COBRA portion of the code is based on a set of two-fluid, three-field, multi-dimensional fluid equations to describe thermal-hydraulic behavior of the reactor vessel component. The TRAC portion of the code is based on a one-dimensional, two-phase drift flux model to describe thermal-hydraulic behavior of the major components of a PWR reactor coolant system, such as SGs, pipes, pumps, valves and pressurizer.

<u>HOTSPOT</u> is used for detailed fuel rod model analysis to calculate the effect of uncertainties at axial locations of the fuel rod.

<u>M-RELAP5</u> is used for the calculation of thermal-hydraulic behavior and safety performance during a small break LOCA. It is based on a non-equilibrium separated two-phase flow thermal hydraulic approach with additional models to describe the behavior of the components of reactor systems, including heat conduction in the core and reactor coolant system structures, reactor kinetics, control systems and trips.

Table 15.0-1 identifies which code or codes were used for analyzing specific events.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

Topical Reports: See Section 15.0.0.2 of this SER.

15.0.2.3 Regulatory Basis

In order to establish a licensing basis, licensees must analyze transients and accidents in accordance with the requirements of 10 CFR 50.34 and 10 CFR 50.46. Guidance set forth in NUREG-0800 Section 15.0.2, "Review of Transient and Accident Analysis Methods" covers the following elements:

- Documentation
- Evaluation model
- Accident scenario identification process
- Code assessment
- Uncertainty analysis
- Quality assurance

15.0.2.4 Technical Evaluation

<u>MARVEL-M</u>

MARVEL-M is the applicant's (MHI's) version of MARVEL, a two-loop simulation code developed by Westinghouse in the 1970s and accepted by the NRC for the design and licensing analysis of specific non-uniform transients for Westinghouse PWR plants. The MARVEL code was licensed to MHI in 1971 and used for operating plant analysis on Japanese PWRs. In the 1990s, MHI expanded the MARVEL code to simulate four coolant loops, added a RCP model and denoted this version as MARVEL–M.

Topical Report (TR) MUAP-07010-P, "Non-LOCA Methodology," describes MARVEL-M, its use in non-LOCA transient analyses and the bases for applying it to the US-APWR. This TR has been submitted to the NRC staff and is currently under review. The staff noted that for each of the six sample transient analyses presented in Section 6 of MUAP-07010-P the timing of events differs from those shown in DCD Chapter 15 for the corresponding analyses. In order to determine if the event-specific findings to be made in the topical report SER are applicable to the DCD analysis, the staff issued RAI 769-5797, Question 15.0.0-25, which requests the applicant explain the differences between the two sets of analyses. In a response dated July 15, 2011, the applicant stated that the DCD was submitted 6 months after the TR; therefore, it includes design changes that were not captured in the TR. The applicant also stated that the purpose of MUAP-07010-P, Section 6, is to provide sample transient analyses, with the DCD providing the licensing basis. The staff finds this acceptable and concludes the event-specific findings made in the SER on MUAP-07010-P can be applied to the DCD analysis.

NRC approval of MUAP-07010-P will be tracked as **Open Item 15.00-4**. Resolution of this open item is required for approval of MARVEL-M for DCD Chapter 15 non-LOCA safety analyses.

VIPRE-01M

VIPRE-01M is the MHI version of VIPRE-01, which was developed by Electric Power Research Institute (EPRI) and generically approved by the NRC for PWR licensing applications as described in NP-2511-CCM-A, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores." The applicant added specific DNB correlations and implemented minor modifications to enable enhanced design application flexibility and denoted this version as VIPRE-01M. MUAP-07009-P, "Thermal Design Methodology," demonstrates that VIPRE-01M is applicable to PWR cores using sensitivity studies, comparisons with other qualified codes and comparisons with DNB test data. This topical report has been submitted to the NRC staff and is currently under review. NRC approval of MUAP-07009-P will be tracked as **Open Item 15.00-5.** The specific use of VIPRE-01M for non-LOCA transient analysis is described in MUAP-07010-P. NRC approval of this report is also being tracked as an open item (described in the preceding paragraph). Resolution of these two open items is required for approval of VIPRE-01M for DCD Chapter 15 non-LOCA safety analyses**.**

TWINKLE-M

TWINKLE-M is the MHI version of TWINKLE, which was developed by Westinghouse and accepted by the NRC for licensing analysis. The TWINKLE code was licensed to MHI in 1971 and used for operating plant analysis on Japanese PWRs. The applicant subsequently expanded the spatial mesh in the code to support three-dimensional core calculations and incorporated a discontinuity factor to improve the accuracy of the local power distribution calculation. The TWINKLE-M code, its use in non-LOCA transient analyses and the bases for applying it to the US-APWR are described in MUAP-07010-P. NRC approval of this TR is being tracked as **Open Item 15.00-4**. Resolution of this open item is required for approval of TWINKLE-M for DCD Chapter 15 non-LOCA safety analyses**.**

RADTRAD

Section 15.0.3 of this SER discusses the application of RADTRAD for DCD Chapter 15 safety analyses.

ANC

Section 4.3 of this SER includes an evaluation of the US-APWR's use of ANC. ANC is applied to non-LOCA methodology as described in MUAP-07010-P. NRC approval of MUAP-07010-P is being tracked as **Open Item 15.00-4**.

WCOBRA/TRAC (M1.0)

WCOBRA/TRAC (M1.0) is the MHI version of WCOBRA/TRAC, which was developed by Westinghouse and approved by the NRC for PWR licensing analysis. It was modified for US-APWR design features and this version was denoted WCOBRA/TRAC (M1.0). MUAP-07011-P, "Large Break LOCA Code Applicability Report for US-APWR," documents WCOBRA/TRAC (M1.0) and provides the basis for US-APWR application. This TR has been submitted to the NRC staff and is currently under review. NRC approval of this TR is being tracked as **Open Item 15.00-6**. Resolution of this open item is required for approval of WCOBRA/TRAC (M1.0) for DCD Chapter 15 large break LOCA safety analyses.

HOTSPOT

WCAP-12945-P-A "Code Qualification Document for Best Estimate LOCA Analysis," documents NRC approval of the HOTSPOT code for best estimate, large-break LOCA analysis. Its use in US-APWR is described in MUAP-07011-P. NRC approval of MUAP-07011-P is being tracked as **Open Item 15.00-6.**

M-RELAP5

M-RELAP5 is the MHI version of RELAP5-3D, which was developed at the Idaho National Laboratory. The applicant modified the code for US-APWR application and this version was denoted M-RELAP5. MUAP-07013-P, "Small Break LOCA Methodology for US-APWR," documents M-RELAP5 and the bases for applying it to the US-APWR. This topical report has been submitted to the NRC staff and is currently under review. NRC approval of this TR is being tracked as **Open Item 15.00-7**. Resolution of this open item is required for approval of M-RELAP5 for DCD Chapter 15 small break LOCA safety analyses.

15.0.2.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.0.2.6 Conclusions

As a result of the open items, the staff is unable to finalize its conclusion on Section 15.0.2 in accordance with the requirements of NRC regulations.

15.0.3 Radiological Consequences of Design-Basis Accidents

15.0.3.1 Introduction

In DCD Tier 2, Chapter 15, the applicant performed radiological consequence assessments of the following seven reactor design-basis accidents (DBAs), using a hypothetical set of atmospheric relative concentration (dispersion) values (χ/Q values) for accidents. Because all other aspects of the design are fixed, these χ/Q values help determine the required minimum distances to the exclusion area boundary (EAB) and the low-population zone (LPZ) for a given site in order to provide reasonable assurance that the radiological consequences of a DBA will be within the siting dose criteria specified in regulation, as identified below.

15.0.3.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: DCD Tier 2 Sections 15.0.3, 15.1.5.5, 15.3.3.5, 15.4.8.5, 15.6.2, 15.6.3, 15.6.5.5, 15.7.4, and 15A provide discussion of the DBA radiological consequences analyses. The DBAs analyzed for radiological consequences are the following:

- Steam system piping failure outside containment (MSLB)

- RCP rotor seizure (locked-rotor accident (LRA))
- Rod ejection accident (REA)
- Failure of small lines carrying primary coolant outside containment
- Steam generator tube rupture (SGTR)
- Loss-of-coolant accident (LOCA)
- Fuel handling accident (FHA)

The applicant provided information on the radiological consequences analysis methodology, assumptions and results for the dose at the EAB, at the LPZ outer boundary, and in the main control room (MCR). The applicant also provided information on the radiological habitability in the US-APWR design technical support center (TSC) to show compliance with the onsite emergency response facility regulatory requirements.

In DCD Tier 2, Chapter 15, the applicant concluded that the US-APWR design will provide reasonable assurance that the radiological consequences resulting from any of the above DBAs will fall within the offsite dose criterion of 0.25 Sievert (Sv) (25 rem) total effective dose equivalent (TEDE), as specified in 10 CFR 52.47(a)(2), “Contents of applications; technical information,” and the control room operator dose criterion of 0.05 Sv (5 rem), as specified in 10 CFR Part 50, Appendix A, GDC 19, “Control Room,” as incorporated by reference in 10 CFR 52.47(a)(3). The applicant reached this conclusion by performing the DBA radiological consequences analyses which:

- use reactor accident source terms based on NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” and RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”;

- credit control of the pH of the water in the containment to prevent iodine evolution;

- use a set of hypothetical atmospheric dispersion factor (χ/Q) values.

The χ/Q values are the relative atmospheric concentrations of radiological releases at the receptor point in terms of the rate of radioactivity release. In lieu of site-specific meteorological data, the applicant provided a reference set of χ/Q values as site parameters for the US-APWR design. The site parameter χ/Q values were selected to envelop a reasonable number of existing nuclear power reactor sites. DCD Tier 2, Table 2.0-1, “Key Site Parameters,” provides the site parameter accident χ/Q values for the US-APWR design, including values for the EAB and LPZ, as well as the accident-specific control room and TSC receptor χ/Q values.

DCD Tier 2, Table 15.0-17, “Summary of Calculated Doses for Events with a Radiological Release,” summarizes the results from the DBA radiological consequence evaluations and compares these results to the applicable dose acceptance criteria.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical/Topical Reports: MUAP-08006-P, "US-APWR Sump Debris Chemical Effects Test Plan"

15.0.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light-Water Reactors," and are summarized below.

1. 10 CFR 50.34(a)(1), as it relates to evaluation and analysis of fission product releases

2. 10 CFR 52.47, "Contents of Applications; Technical Information," paragraph (a)(2), as it relates to the evaluation and analysis of the offsite radiological consequences of postulated accidents with fission product release.

3. 10 CFR Part 52.47(a)(2)(iv), as it relates to ability of plant systems to mitigate the radiological consequences of plant accidents.

4. GDC 19, as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.

5. 10 CFR 100.21, "Non-seismic Siting Criteria," as it relates to the evaluation and analysis of the radiological consequences of postulated accidents for the type of facility to be located at the site in support of evaluating the site atmospheric dispersion characteristics.

6. 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Paragraph IV.E.8, as it relates to adequate provisions for an onsite TSC, from which effective direction can be given and effective control can be exercised during an emergency.

7. GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," as it relates to isolation of all pipes that are part of the RCPB and penetrate the containment building.

Review interfaces with other SRP sections can also be found in NUREG-0800, Section 15.0.3. The staff also referred to NUREG/CR-5950, "Iodine Evolution and pH Control" in this area of review.

The related acceptance criteria are as follows:

1. Regulatory Guide 1.183
2. SRP Section 6.5.2 (NUREG-0800) contains detailed requirements for evaluation of water pH in plants that employ containment sprays.

15.0.3.4 Technical Evaluation

The staff evaluated the calculated radiological consequences of DBAs against the dose criteria, given in 10 CFR 52.47(a)(2)(iv), of 0.25 Sv (25 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release, and 0.25 Sv (25 rem) TEDE at the outer boundary of the LPZ for the duration of exposure to the release cloud. The staff used a

criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from DBAs in the control room of the US-APWR design, pursuant to GDC 19. The staff used applicable guidance in SRP Section 15.0.3 and RG 1.183 in its review of the US-APWR DBA radiological consequence analyses. Although RG 1.183 applies to the current operating power reactors, its guidance on radiological acceptance criteria, formulation of the source term, and DBA modeling is useful in the review of the US-APWR design.

The staff evaluated the DBA radiological habitability analysis for the US-APWR design TSC against the onsite emergency response facility regulatory requirements in 10 CFR 50, Appendix E, Paragraph IV.E.8, and 10 CFR 50.47(b)(8) and (b)(11), "Emergency Plans." The staff's complete review of the emergency response facilities is discussed in Section 13.3, "Emergency Planning," of this SER.

The staff reviewed the radiological consequence analyses performed by the applicant using the hypothetical χ/Q values given in DCD Tier 2, Table 2.0-1 as site parameters. The staff finds that the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria stated above. To evaluate the applicant's analyses, the staff performed independent radiological calculations for the above DBAs using the site parameter χ/Q values provided by the applicant and the RADTRAD, Version 3.03, computer code described in Supplement 2 to NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation." The following sections describe the staff's findings.

The applicant followed the accident analysis guidance in RG 1.183 and SRP Section 15.0.3. The US-APWR DBA radiological consequences analyses credit safety-related SSCs for mitigation of the radiological consequences of a DBA. The analyses evaluated the DBAs considering a single, active failure that maximizes the radiological consequences and additionally assume a loss of offsite power (LOOP), if the case with LOOP is found to be limiting. Each analysis conservatively assumed the intake flow corresponding to two train actuation and the recirculation flow corresponding to one train actuation of the emergency filtration systems for the main control room for the duration of the accident.

15.0.3.4.1 Accident Source Terms

The US-APWR is an evolutionary PWR design. The primary system design and main components design are similar to those of currently operating reactors, and the plant design includes active ESFs to mitigate accidents. In SECY-94-302, "Source Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs," dated December 19, 1994, the staff proposed to use only the coolant, gap, and early in-vessel releases from NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" for the radiological consequence assessments of DBAs for evolutionary and passive light-water reactor (LWR) designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These scenarios define the most severe accidents from which the plant could be expected to return to a safe-shutdown condition. The revised source terms in NUREG-1465 must be applied conservatively in evaluating DBAs in conjunction with conservative assumptions in calculating doses, such as adverse meteorology. Application to severe accidents may use more realistic assumptions.

The staff considered the inclusion of the ex-vessel and the late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive

LWRs, the estimated frequencies of such scenarios are low enough that they do not have to be considered credible for the purpose of meeting 10 CFR 50.34, as reiterated in 10 CFR 52.47. In its Staff Requirements Memorandum related to SECY-94-302, the Commission approved the staff-recommended technical positions to use only the coolant, gap, and early in-vessel releases in NUREG-1465 for the radiological consequence assessments of DBAs for evolutionary and passive LWR designs.

The NRC issued RG 1.183 in July 2000 to provide guidance to licensees of operating power reactors on acceptable applications of alternative source terms pursuant to 10 CFR 50.67, “Accident Source Term.” This RG provides guidance based on insights from NUREG-1465 and significant attributes of other alternative source terms that the NRC staff may find acceptable for operating LWRs. It also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted alternative source term for operating power reactors. In SRP Section 15.0.3, the staff’s review procedures direct the use of RG 1.183 regulatory positions, as far as applicable to the plant design under review. The applicant followed the relevant guidance in RG 1.183 for PWRs.

For DBAs other than the LOCA and the FHA, the source of radioactive materials available for release can be the primary and secondary coolant. The staff’s review of the coolant source terms is discussed in Section 11.1 of this SER.

The iodine appearance rate is used as a basis for input to the MSLB, SGTR and small line break accident radiological consequences analyses, with regard to the assumptions on iodine spiking in the coolant. In RAI 38-412, Question 15.00.03-2, the staff requested that the applicant provide the calculation of the iodine appearance rates listed in DCD Tier 2, Table 15.0-11, “Iodine Appearance Rates in the Reactor Coolant (Ci/min)” including the basis for the inputs and assumptions and explain why the iodine appearance rate varies between the three accidents identified in the table.

The applicant’s response, dated August 22, 2008, described an iodine appearance rate calculation model different from what the staff has seen in previous submittals. In RAI 105-1624, Question 15.00.03-24, the staff requested additional information on the iodine appearance rate calculations and justification for differences from the guidance in RG 1.183, specifically with regard to assumptions on modeling the equilibrium coolant concentration as dose equivalent iodine-131 (DEI-131), radioactive decay, leakage, coolant cleanup, and transient conditions.

By letter dated January 6, 2009, the applicant responded with additional information to describe the differences in the accident-specific assumptions and show that modeling of iodine spiking used in the US-APWR analyses is bounding for the usual model seen by the staff and is also bounding for the accident conditions expected for MSLB, SGTR and small line break accident. The applicant also clarified that committed effective dose equivalent (CEDE) dose conversion factors were used in the adjustment of the coolant concentration to DEI-131.

The staff developed iodine spiking source terms using the RG 1.183 model, and compared the resulting accident-related iodine values to those calculated by the applicant. The staff’s calculated iodine spiking release rates for I-131 are bounded by the applicant’s values. The spiking release rates for I-133 and I-135 are approximately equivalent; while the staff’s values for I-132 and I-134 bound the applicant’s values, but within the level of uncertainty for the calculation. To verify that the applicant’s modeling of iodine spiking does not adversely affect the estimate of the radiological consequences of the MSLB, SGTR and small line break

accident, the staff performed independent analyses of these DBAs using the RG 1.183 iodine spiking model assumptions and confirmed the applicant's dose results. Based on the applicant's response and the staff's independent assessment, the staff finds that the applicant has sufficiently addressed the staff's questions on coolant iodine spiking. Therefore, the staff considers **RAI 38-412, Question 15.00.03-2**, and **RAI 105-1624, Question 15.00.03-24, resolved**.

The US-APWR DBA radiological consequences analyses are based on 102 percent of rated core thermal power. The core fission product inventory for use in the DBA radiological consequences analyses is given in DCD Tier 2, Table 15.0-14, "Reactor Fission Product Nuclide Inventory and Related Parameters," and is repeated in DCD Tier 2, Table 15A-10. The applicant calculated the core fission product isotopic inventory at 102 percent of the core rated thermal power of 4451 megawatts-thermal (MWt) (i.e., 4540 MWt). The applicant used the ORIGEN-2.2 isotope generation and depletion computer code along with extended burnup libraries for ORIGEN 2 high burnup reactor models to calculate the core isotopic inventory. RG 1.183 states that ORIGEN 2 is an appropriate code for calculation of the core inventory; however, the staff had questions related to its use as discussed below.

The staff noted that the ORIGEN 2.2 generation and depletion code was used to calculate the core radionuclide inventory. Oak Ridge National Laboratory (ORNL) does not support the ORIGEN 2 code any longer, but instead recommends use of the ORIGEN-ARP or ORIGEN-S code included in the SCALE code package, which is kept up-to-date. In RAI 38-412, Question 15.00.03-20, the staff requested that the applicant justify the use of an older unsupported version of the ORIGEN code. SCALE 5.1 was the latest release at the time of the RAI, and includes libraries for high burnup fuel, up to 72 gigawatt days per metric ton uranium (GWD/MTU).

By letter dated October 20, 2008, the applicant provided an evaluation comparing the ORIGEN-2.2 core inventory against an ORIGEN-ARP core inventory for the US-APWR core. For the ORIGEN-ARP calculation, the core inventory was calculated using the same conditions for burnup, enrichment, power and fuel cycle assumptions as were used in the DCD calculation using ORIGEN-2.2. The applicant compared the inventories calculated by each version of the ORIGEN code for the top 20 nuclides that account for 99 percent of the total LOCA dose. The applicant's evaluation shows that the difference between the inventories calculated by each version of the code is approximately 1 percent, with a consequent difference between the doses based on the inventories estimated to be 1 percent. This difference is within the uncertainty for the overall dose analysis, and shows that the use of the ORIGEN-2.2 code is appropriate.

DCD Tier 2, Subsection 15A.1.1.3 states that the fuel burnup is 55 GWD/MTU in two cycles. In RAI 38-412, Question 15.00.03-21, the staff requested that the applicant confirm that the cross-section libraries used in the calculation of the core fission product inventory are applicable to the maximum fuel burnup assumed. By letter dated October 20, 2008, the applicant responded by affirming that the cross-section libraries for extended burnup PWR fuel, extrapolated to 55 GWD/MTU, were used in the calculation of the core inventory. In the ORIGEN-ARP calculation provided in the RAI response, the applicant used the W17x17 cross-section library, which is applicable up to 72 GWD/MTU. The staff performed some limited confirmatory analyses with ORIGEN-ARP using the extended burnup inventories up to 72 GWD/MTU and varying the cycle lengths and was able to determine that the cross-section libraries and assumptions on core operation used by the applicant in the core inventory calculations were acceptable.

Based on the responses to RAI 38-412, Questions 15.00.03-20 and -21, the staff finds that the applicant has sufficiently addressed the staff's questions on use of the ORIGEN-2.2 code. Based on the above discussion, the staff finds that the applicant's calculation of the core fission product isotopic inventory conforms to RG 1.183 guidance and is acceptable. Therefore, the staff considers **RAI 38-412, Questions 15.00.03-20 and 21, resolved**.

15.0.3.4.2 Hypothetical Atmospheric Dispersion Factors

Because no specific site is associated with the US-APWR plant, the applicant defined the offsite boundaries only in terms of site parameters which consist of hypothetical atmospheric relative concentration (χ/Q) values at fixed EAB and LPZ distances. DCD Tier 2, Tables 2.0-1 and 15A-17 through 15A-23, list the accident χ/Q values used in the radiological consequence analyses for the US-APWR design. Section 2.3.4 of this report provides discussion of the staff's review of the site parameter atmospheric dispersion factors. The staff will perform an independent assessment of the site characteristic short-term (less than or equal to 30 days) atmospheric dispersion factors for accident consequence analyses for a COL application that references the US-APWR design. If the site characteristic atmospheric dispersion factors exceed the site parameter values used in this evaluation (i.e., poorer dispersion characteristics), a COL applicant may have to consider compensatory measures, such as increasing the size of the site or providing additional ESF systems to meet the relevant dose limits set forth in 10 CFR 52.79 and GDC 19. The US-APWR DCD includes the following COL information item to address this possibility.

COL 15.0 (1) states:

> *"In the COLA* [COL application]*, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and 10 CFR 52.79 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criterion 19 are met for affected events using site-specific χ/Q values. Additionally, the Technical Support Center (TSC) dose should be evaluated against the habitability requirements in Paragraph IV.E.8 to 10 CFR Part 50, Appendix E, and 10 CFR 50.47(b)(8) and (b)(11)."*

The staff agrees with the intent of the COL item which identifies additional information needed in the COLA to address the case where the DCD is not bounding.

15.0.3.4.3 Independent Calculation of Containment Water pH in Design-Basis Accident

Management of the post-accident containment water chemistry must comply with the requirements of GDC 41, "Containment Atmosphere Cleanup," and GDC 4, "Environmental and Dynamic Effect Design Bases." By minimizing the release of radioactive iodine from the containment sump water, the water chemistry will meet the requirement of GDC 41, as it relates to the ability of the design of containment atmosphere cleanup systems to control fission product releases to the reactor containment following postulated accidents. By preventing stress-corrosion cracking of stainless steel components exposed to the water accumulated in the containment sump, the water chemistry will meet the requirement of GDC 4 that components important to safety be compatible with the environmental conditions associated with accident conditions, including LOCAs.

NUREG-1465 states that, after an accident, iodine entering the containment from the reactor core is composed of at least 95 percent cesium iodide (CsI), with the remaining 5 percent elemental iodine and a small amount of hydriodic acid. However, about 3 percent of elemental iodine in contact with some organic compounds will produce organic iodides. Therefore, the iodine in the containment will consist of 95 percent particulate iodine as CsI, 4.85 percent elemental iodine (I_2), and 0.15 percent organic iodine. The composition of the iodine in the US-APWR is consistent with the composition stated in NUREG-1465.

Iodine in the form of CsI is soluble in the containment water. However, some of it may be converted into the elemental form, which is considerably less soluble, and will be released into the containment atmosphere. The released radioactive iodine may leak out of the containment and contribute to outside radiation doses. To minimize formation of the elemental iodine, the pH of the containment water should be kept basic. Basic pH will also prevent stress-corrosion cracking of the stainless steel components.

15.0.3.4.3.1. Regulatory Criteria

The general acceptance criterion for maintaining pH in containment spray systems, sumps and pools is based on:

1. 10 CFR 50.34(a)(1), paragraph (a)(1), as it relates to evaluation and analysis of fission product releases.

2. 10 CFR 52.47(a)(2)(iv), as it relates to ability of plant systems to mitigate the radiological consequences of plant accidents.

Implementation of these criteria are accomplished by satisfying RG 1.183, “Alternative Radiological Source Terms.” In addition, SRP Section 6.5.2 contains detailed guidance for evaluation of water pH in plants that employ containment sprays.

15.0.3.4.3.2. Summary of Technical Information

The principal repository of iodine in DBA scenarios is the refueling water storage pit (RWSP), which is described briefly in DCD Section 6.3.2.2.3. Normally, the RWSP water contains boric acid (4200 parts-per-million boron (ppm B)), which lowers pH to about 4.5 at room temperature.

Under accident conditions, the primary pH control chemical for iodine retention is sodium tetraborate decahydrate (NaTB), commonly known as borax. At least 44,100 pounds (lb) (20,000 kilograms (kg)) of this dry chemical resides along the circular periphery of the containment, in 23 baskets (DCD Section 6.3.2.2.5). These baskets are specially designed to catch spray water, which dissolves the NaTB and then drains to the RWSP, increasing its pH. The applicant claims that the NaTB ensures a pH of at least 7. However it also mentions that it takes 12 hours for all the NaTB to dissolve (DCD Section 6.3.2.2.5). No actual calculation of pH is provided in DCD Sections 15.0.3, 6.3.2, 6.5.2, or any other section, although the applicant has provided a pH calculation in its responses to various RAIs, as explained in detail in the following section.

DCD Section 15.6.5.5.1.1, supplemented by additional information supplied by the applicant response to RAI 176-1987 dated March 3, 2009, provides specific assumptions about contents of water that leaves the primary system for the containment in a DBA:

1) Fission products iodine and cesium have total inventories of 340 mol and 3900 mol, respectively. These amounts include all isotopes, both stable and radioactive, since the entire elemental amounts contribute to the pH calculation.

2) All primary system water enters the RWSP, together with TS limits of iodine. Compared to the iodine mentioned in 1) above, this amount of iodine is negligible, and thus will be ignored. This statement implies the concentrations in primary system water listed in DCD Table 5.2.3-2, "Recommended Reactor Coolant Water Chemistry Specification." From the applicant's response to RAI 176-1987, Question 27 dated March 3, 2009, the total RCS water volume is 134,730 gallons (510 cubic meters (m^3)). Nominal values of boric acid and LiOH are given in [DCD] Table 5.2.3-2 as 4000 ppm and 0.2 ppm, respectively. However, additional information supplied by the applicant in MUAP-08006-P, Revision 1, suggests that absolute limits are 4200 ppm B (maximum) and 0 ppm Li (minimum). These values will be used, since they are likely to result in a lower (conservative) calculated pH.

Water from the RCS mixes with the water in the RWSP, which contains a nominal volume of 81230 cubic feet (ft^3)(2300 m^3).

The calculation of pH requires knowledge of all possible constituents in the water, consistent with the requirements in SRP Section 6.5.2 ("Review Procedures," III.4.C.ii, p. 6.5.2-12). It is possible that acids produced by radiolysis in containment may lower the pH over time, and these should be considered for completeness:

1) The formation of nitric acid (HNO_3) by radiolysis of water-air mixtures. The applicant supplied a table of values for this quantity, which is duplicated in the table below.

Table 15.0.3.3-1 Dose to Containment Airspace and Water During Accident Sequence

Time	Cumulative Absorbed Dose (kGy)	
(h)	Cable Jacketing	Containment water
0.1	26	2.5
0.2	52	4.9
0.3	76	7.1
0.4	100	9.2
0.5	120	11
1	230	19
2	400	32
3	530	41
5	730	56
10	1100	87
20	1500	130
30	1800	160
50	2300	200
70	2600	240
100	3100	280
200	4200	390
300	4900	480
400	5500	560
500	5900	620

600	6300	680
720	6600	740

2) The formation of hydrochloric acid (HCl) by irradiation of electrical cable jacket and insulation materials. This source is gaseous, but would quickly and easily be washed out of the air by sprays. This model requires knowledge of the amount of cable insulation, temperature, and dose rate to cable insulation. The applicant has estimated the total amount of cable jacketing as 6000 kg, and supplied the dose rates as represented in the table above. It recommends evaluation at three temperatures: 50, 100, and 150 °C.

15.0.3.4.3.3. Technical Assessment

15.0.3.4.3.3.1 Background

This assessment (evaluation) is performed in conjunction with DCD Section 15.0.3, which does not specifically mention such a pH calculation. However, it is performed under the general stipulations that the "…staff's evaluation may include verification that the applicant followed applicable guidance, performance of independent calculations, and/or validation that the appropriate assumptions were made." [SRP Section 15.0.3, p. 15.0.3-15]. Such a calculation would be generally needed to meet the requirements of SRP Section 15.0.3 Acceptance Criterion 1 [based on 10 CFR 50.34(a)(1)], which relates to evaluation and analysis of fission product releases.

A pH calculation is also important to the evaluation of SRP Section 6.5.2, SRP Acceptance Criterion 1.G, by which water containing iodine must have a pH of 7, in order to avoid revolatilization of the iodine. This section also provides that the possibility of iodine re-evolution must be considered (Review Procedures 1.E, p. 6.5.2-9), and that all possible solutes that would affect pH should be included (Review Procedures 4.C.ii, p. 6.5.2-12). The applicant has stated that its fission product cleanup system will ensure pH of 7 (DCD section 6.5.2), and supporting calculations were supplied in response to RAI 460-3484 dated November, 13, 2009.

15.0.3.4.3.3.2. Preliminary Evaluation

As a preliminary step, the staff evaluates the simple system formed when all primary system water is added to the RWST, and assumes that all NaTB is instantly dissolved. Upon dissolution in water, NaTB hydrolyzes to become boric acid and the strong base sodium hydroxide (NaOH):

$$Na_2B_4O_7{\bullet}10H_2O \rightarrow 4\ H_3BO_3 + 2\ NaOH + 3\ H_2O. \qquad (1)$$

The NaOH dissociates completely, and the hydroxide ions neutralize the boric acid, which raises the pH of the solution. (Boric acid is a weak acid, and only a small amount of ionization occurs; hence, the available base can easily neutralize the small amount of acid from both the NaTB and the boric acid already in water.) From Eq. (1), each mole of NaTB dissolved produces four additional moles of boric acid and two of NaOH (and releases a small amount of additional water, which is negligible in the calculation). The concentrations for boric acid and base are based on the quantities given in the table below, which provides a conservative estimate using the maximum values for water volume and acid. These concentrations are

converted to molality (mol solute per kg water) since these units are required by the pH calculation. Calculations of various concentrations are straightforward, and given below:

Table 15.0.3.4.3-2
Parameters for pH Calculation

Quantity	Value	Reference
RCS water volume	134,730 gal.	Response to RAI 176-1987
RWST water volume	81230 ft^3	DCD 6.3.2.2.3
Boric acid in RCS	4000 ppm	DCD Table 5.2.3-2
LiOH in RCS	0.0 ppm	DCD Table 5.2.3-2
Boric acid in RWST	4200 ppm	MUAP-08006-P (Rev. 1)
Normal temperature in RWST	70-120°F	DCD 6.3.2.2.3
NaTB in containment	44100 lb	DCD 6.3.2.2.5
CsOH (fission product) in water	1068 mol	Response to RAI 416-2916

- Total water (RCS + RWST) = 2810 m^3 ≈ 2.81 x 10^6 kg
- Total NaTB = 44,100 lb = 20,000 kg = 52443 mol
- Nominal concentration of boric acid = (4000 ppm B) = 0.372 *m*
- Nominal concentration of LiOH in RCS = 0.2 ppm = 8.4 x 10^{-6} *m*
- Concentration of LiOH in (RCS+RWST) water = 1.5 x 10^{-9} *m*
- Concentration of boric acid due to NaTB dissolution = 0.075 *m*
- Concentration of NaOH due to NaTB dissolution = 0.037 *m*

Both LiOH and NaOH are strong bases that will dissociate virtually completely; their behavior is so similar that they can be treated as the same substance. The concentration of NaOH from NaTB dissolution dwarfs the amount of LiOH added from the RCS. Furthermore, the applicant, in MUAP-08006-P, suggests that the lower limit for LiOH in the RCS is zero; hence, none will be considered. Thus, a pH calculation can be based on the combined amounts of boric acid (0.446 *m* ≈ 4800 ppm B) and the NaOH added by the NaTB dissolution (0.037 *m*).

The EPRI Pressurized Water Chemistry Guidelines[3] (Appendix A) provide a methodology for calculating pH that involves solving simultaneous chemical equilibria. The equations involve boric acid polymerization, water dissociation, and ionic strength effects, and are based on published literature. These equations were solved for the system described above, and the results given in Table 15.0.3-3. From the table it can be seen that the dissolution of NaTB does indeed raise the pH above 7 at both room temperature and at the elevated temperature representing accident conditions.

Table 15.0.3.4.3-3
Calculation of pH in RWST

Description	B	LiOH or NaOH (*m*)	T (K)	pH
RWST normal operation	4000 ppm	0	300	4.53
RWST + RCS accident	4800 ppm	0.0373	300	7.67
RWST + RCS accident	4800 ppm	0.0373	373	7.78

This analysis does not include the effects of fission products, which would likely raise the pH, or radiolytic acids, which would lower pH.

15.0.3.4.3.3.3. Complete analysis

As noted in DCD Section 6.3.2.2.5, the full dissolution of all NaTB takes 12 hours, which suggests that the pH rises gradually over this time period. Since fission products are released at the beginning of the accident, radiation doses to air and water in the containment start the radiolytic generation of acids at the start of the accident. Thus, the pH in the RWST may be decreased initially due to acid additions, and it is possible that it will remain below 7 for a significant interval. In this complete analysis, all relevant effects on pH are considered, as outlined below.

Fission Products. Fission product iodine is assumed to combine with fission product cesium to form the salt CsI. However, this consumes less than 10 percent of the cesium, and the remainder is presumed to be CsOH, which is added to water as a strong base. The applicant has noted in its response to RAI 176-1987 that total core inventories of these elements are 340 mol and 3900 mol, respectively. For a DBA, only 30 percent of the cesium is assumed to be released, resulting in the entry for CsOH in Table 15.0.3.4.3-2.

Nitric acid. Radiation dose to air-water systems has been demonstrated to produce nitric acid, which lowers the pH of the water.[3,4] An acid generation rate that is well established is:

$$G(HNO_3) = 0.007 \text{ molecules } HNO_3 / 100 \text{ eV dose.} \quad (2)$$

Using the water-dose information supplied by the applicant (Table 15.0.3.3-1, column 3), the cumulative inventory of HNO_3 is calculated using Eq. (2) and shown in Table 15.0.3.4.3-4, column 2.

Table 15.0.3.4.3-4
Radiolytic Acid Generation and Dissolution of NaTB in Containment

Time (h)	HNO_3 (mol)	HCl (mol)	Total Acid[a] (*m*)	Net Base[b] (*m*)	H_3BO_3 (*m*)
0.1	5	100	3.75E-05	0.000274	0.388427
0.2	10	200	7.49E-05	0.000547	0.389049
0.3	14	293	1.09E-04	0.000824	0.389671
0.4	19	385	1.44E-04	0.0011	0.390293
0.5	22	462	1.73E-04	0.001383	0.390916
1	39	886	3.29E-04	0.002781	0.394026
2	65	1541	5.72E-04	0.005649	0.400247
3	84	2042	7.57E-04	0.008575	0.406468
5	114	2813	1.04E-03	0.014511	0.41891
10	177	4239	1.57E-03	0.029533	0.450015
20	265	5780	2.15E-03	0.035174	0.462457
30	326	6936	2.58E-03	0.034741	0.462457
50	408	8863	3.30E-03	0.034026	0.462457
70	489	10019	3.74E-03	0.033586	0.462457

Time (h)	HNO_3 (mol)	HCl (mol)	Total Acid[a] (*m*)	Net Base[b] (*m*)	H_3BO_3 (*m*)
100	571	11946	4.45E-03	0.032871	0.462457
200	795	16185	6.04E-03	0.031283	0.462457
300	979	18883	7.07E-03	0.030258	0.462457
400	1142	21195	7.95E-03	0.029377	0.462457
500	1264	22736	8.54E-03	0.028785	0.462457
600	1386	24278	9.13E-03	0.028193	0.462457
720	1509	25434	9.59E-03	0.027738	0.462457

[a] Total strong acid = HNO_3 + HCl
[b] Net base = NaOH + LiOH – Total strong acid

Hydrochloric acid. The initial work on this effect was conducted by Wing[4], and then developed for reactor safety applications by Beahm and coworkers.[2] Using several approximations, it can be distilled to the simple form:

$$R = (1.32 \times 10^{-15}\ E_{□} + 8.70 \times 10^{-16}\ E_{□})\ W / V, \qquad (3)$$

where R is the production rate of HCl (mol/s), $E_{□}$ and $E_{□}$ are the total energy release rates by □ and □ radiation (in mega-electron-volts/second (MeV/s)), W is the weight of cable insulation (lb), and V is the airspace volume in containment (cubic centimeters (cm^3)). The weight of cable insulation was supplied by the applicant[1] as 6000 kg (13,200 lb.). The volume of containment is given as $2.8 \times 10^6\ ft^3 = 7.93 \times 10^{10}\ cm^3$ (DCD Table 6.2.1-5). The applicant supplied a total containment dose rate that includes both □ and □ sources (Table 15.0.3-1, column 2). It is conservative to assume that the entire dose is due to □ radiation, since this is the higher term in Eq. (3). Using these values, the cumulative production of HCl in containment is given in Table 15.0.3.4.3-4, column 3.

Complete calculation. This calculation is similar to the one presented in Section 15.0.3.4.3.2 of this SER. However, in place of the model from Appendix A of the EPRI Pressurized Water Chemistry Guidelines[6], the staff used the updated equilibrium expressions from "Boric Acid Hydrolysis: A New Look At The Available Data," by Palmer, Benezath, and Wesolowski, 2000, together with the activity coefficient representation from "J. Soln. Chem," by Sweeton, Mesmer, and Bae, 1974. Again, LiOH and NaOH are considered the same species, since both are strong bases. The addition of NaTB is done over the first 12 h, assuming a uniform rate of addition. This increases the concentrations of both boric acid and strong base, as shown in columns 5-6 of Table 15.0.3-4. The NaOH is assumed to completely neutralize any strong acid present (i.e., HNO_3 and HCl). The excess base (after neutralization) is shown in column 5 of the table. This amount, together with the total boric acid, is used as input to the pH calculation.

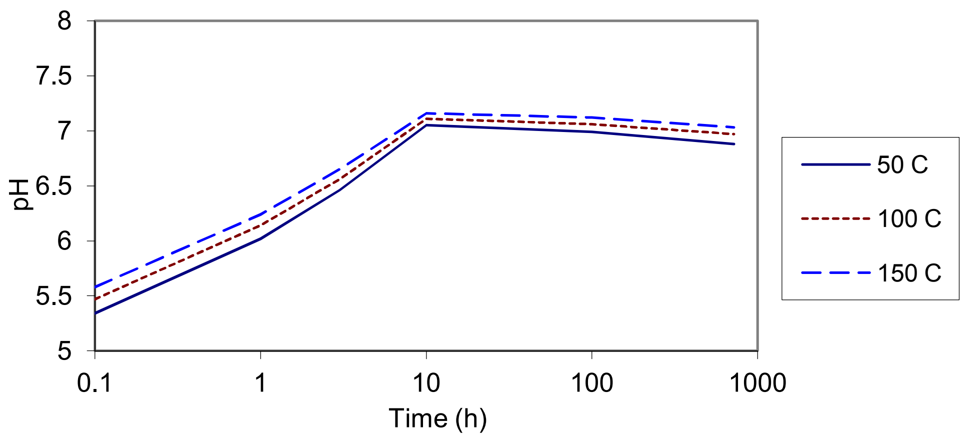
The pH is calculated for three different temperatures (50, 100, and 150 °C), and at times 0.1, 1, 3, 10, 100, and 720 hours. These results are shown in Table 15.0.3.4.3-5 and Figure 15.0.3.4.3-1.

Table 15.0.3.4.3-5
Calculated pH Values

Time (h)	pH 50°C	pH 100°C	pH 150°C
0.1	5.34	5.47	5.58
1	6.02	6.14	6.24
3	6.46	6.56	6.65
10	7.05	7.11	7.16
100	6.99[a]	7.06	7.12
720	6.88[b]	6.97[c]	7.03

[a] Calculation using activity coefficient representation from EPRI Guidelines gave pH=7.15
[b] Calculation using activity coefficient representation from EPRI Guidelines gave pH=7.04
[c] Calculation using activity coefficient representation from EPRI Guidelines gave pH=7.15

Figure 15.0.3.4.3-1
Calculated pH in Containment Water



As expected, the pH rises steadily for the first 12 hours, while the NaTB is being added. It declines slightly through the rest of the transient due to the addition of radiolytic acids. At all three temperatures, the pH rises above 7 within the first 10 hours, and remains barely above 7 through most of the accident sequence. A few values do dip slightly below pH=7 near the end of the sequence, although the EPRI model expression produces values slightly higher, as shown in Table 15.0.3.4.3-5. It is not possible to state which of these values is “best”, since their deviation from each other is a good measure of the uncertainty in the calculation itself. However, the calculations indicate that there is very little cushion, as pH is never very far above 7. This is also true of the applicant’s own calculations (response to RAI 460-3484, Question 06.05.02-7), which barely exceed pH=7 as well.

15.0.3.4.3.4. Conclusions

This independent calculation indicates that pH in the RWST does indeed rise above 7 during a DBA sequence, and remain there through 30 days within the uncertainty of the calculation. These results suggest that pH may be below 7 for several hours after the start of the accident, and this value assumed that NaTB additions began at the start of the accident. Thus, for the first several hours, the pH will be below 7, and the containment sump may be unable to prevent revolatilization of iodine. (This issue of timing is dealt with more thoroughly in the review of

DCD Section 6.5.2.) The NaTB is effective in raising pH, and countering the acidity of the boric acid already in water and the radiolytic acid additions that occur throughout the sequence.

Therefore, the staff concludes that the mitigation effects of a pH greater than 7 will be met for water in the RWSP and closely connected volumes, as well as the containment sprays, for the duration of an accident sequence except for the first few hours. Hence, the applicant has demonstrated compliance with 10 CFR 50.34(a)(1) and 10 CFR 52.47(a)(2)(iv), in conjunction with RG 1.183 and relevant sections of the SRP. This review is based on information supplied by the applicant, the staff review of that information, and on the staff's independent calculation of pH using that information.

15.0.3.4.4 Radiological Consequences of Main Steam Line Break Outside Containment

The applicant has evaluated the radiological consequences of a postulated MSLB accident occurring outside of the containment with failure of one MSIV. The applicant submitted a radiological analysis for the MSLB accident in DCD Tier 2, Subsection 15.1.5.5. DCD Tier 2, Section 15.1.5, discusses the nuclear steam supply system (NSSS) response analysis for a spectrum of steam system piping failures inside and outside of containment. For the purposes of the radiological consequence analysis, the limiting MSLB is a double-ended guillotine break of a main steam line with failure of the MSIV. A limiting failure of one MSIV to close is assumed so that one SG blows down through the break.

The bounding case steam release used in the radiological consequence analysis corresponds to the hot zero-power case with offsite power unavailable. The radiological consequence analysis assumes that the reactor is cooled by releasing steam from the SGs in the unaffected RCS loops. After the break, the SG in the faulted loop is isolated and allowed to steam dry, with the release assumed to be directly to the environment. Iodine released from intact SGs through primary-to-secondary leakage is assumed to mix in the secondary coolant and be partitioned between liquid and steam phases before being released through the main steam safety valves (MSSVs) or main steam relief valves (MSRVs). Noble gases entering the secondary coolant are assumed to be released directly to the environment. The steam releases through the unaffected SGs end in fourteen hours when the RHRS initiates.

The core response analysis precludes fuel damage from occurring as a result of a steam line break; therefore the source of radioactive material for release is the reactor coolant. The applicant analyzed this hypothetical accident for two coolant source term cases.

For the accident-induced iodine spiking case, the analysis assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the MSLB. Before the postulated accident, the US-APWR reactor was assumed to operate at the TS 3.4.16 equilibrium iodine concentration limit of 37 kilobecquerel per gram (kBq/gm) (1.0 microCuries/gram (µCi/gm)) for DEI-131 and 11.1 MBq/gm (300 µCi/gm) for DE Xe-133 in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500, resulting in a rising iodine concentration in the primary coolant during the course of the accident.

For the pre-accident iodine spiking case, the analysis assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum instantaneous concentration limit of 2.2 MBq/gm (60 µCi/gm) for DEI-131, specified in the US-APWR TS. The

staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183 guidance, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the two coolant activity scenarios for the MSLB accident. The staff's analyses followed the guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis MSLB is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated MSLB accident with accident-induced iodine spiking will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.47.

The staff also concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated MSLB accident with pre-accident iodine spiking in the coolant will not exceed the dose criterion set forth in 10 CFR 50.47 (i.e., 0.25 Sv (25 rem) TEDE).

The applicant stated that the doses in the MCR and TSC for the MSLB are bounded by the doses calculated for the LOCA, and provided dose estimates for the MCR. Modeling of the TSC is not discussed in detail in DCD Tier 2, Section 15.1.5.5, nor are TSC dose results for the MSLB provided. However, the dose in the TSC from DBAs is discussed in DCD Tier 2, Subsection 15.6.5.5.1.3 as being represented by the MCR consequences. The staff's review of the TSC habitability is discussed in further detail below in Section 15.0.3.4.11 of this SER.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the MSLB, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for each of the MSLB iodine spiking cases. Therefore, the staff has determined that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis MSLB meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.5 Radiological Consequences of Reactor Coolant Pump Rotor Seizure

The applicant submitted a radiological analysis for the RCP rotor seizure or locked-rotor accident (LRA) in DCD Tier 2, Section 15.3.3.5. The LRA assumes instantaneous seizure of an RCP rotor, which leads to a reactor trip. The radiological consequences analysis assumes that offsite power is unavailable. The source of the radioactive material is the reactor coolant and any release from damaged fuel rods in the core to the primary coolant. The LRA dose analysis is bounding for the radiological consequences of a postulated RCP shaft break, as noted in DCD, Tier 2, Section 15.3.4.

The applicant analyzed this hypothetical accident assuming that 10 percent of the fuel rods in the core will experience local clad temperatures that exceed limits and fail, releasing the entire fission product inventory in the fuel-cladding gap of these rods to the reactor coolant. This fuel-failure assumption bounds the number of rods predicted to fail in the DNB analysis. The primary coolant concentration is assumed to be at the US-APWR proposed TS equilibrium value of 37 kBq/gm (1 μCi/gm) for DEI-131 and 11.1 MBq/gm (300 μCi/gm) for DE Xe-133. The maximum allowable 0.394 liters/minute (l/min) (150 gallons per day (gpd)) of primary-to-secondary leakage through any SG, as specified in the US-APWR TS, carries the activity released to the primary coolant into the secondary coolant. The activity in the primary coolant is transferred to the secondary coolant through primary-to-secondary leakage, with iodine portioning between the liquid and steam in the SG secondary side. The activity is released to the environment until 14 hours after the RCP rotor seizure, at which time the plant cooldown is switched from SG steaming to the RHRS.

The US-APWR SG is designed for a maximum moisture carryover of 0.1 percent, as described in DCD Tier 2, Table 5.4.2-1. In accordance with RG 1.183, Appendix E, Section 5.5.4, the applicant assumed particulate retention on the SGs based on the design moisture carryover. Noble gas and iodine transport through the SGs is considered separately. Because these assumptions conform to the guidance in RG 1.183, the staff finds them acceptable.

The staff has reviewed the applicant’s analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183 guidance, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant’s assessment, the staff performed independent radiological consequence calculations for the LRA. The staff’s analyses followed guidance in RG 1.183 and used the applicant’s assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the LRA.

Based on the comparison of the applicant’s analysis methodology to the guidance in RG 1.183 and the staff’s confirmatory analysis, the staff finds that the applicant’s analysis of the design basis LRA is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated reactor primary coolant pump seizure accident will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.47.

The applicant stated that the doses in the MCR and TSC for the LRA are bounded by the doses calculated for the LOCA, and provided dose estimates for the MCR. Modeling of the TSC is not

discussed in detail in DCD Tier 2, Section 15.3.3.5, nor are TSC dose results for the LRA provided. However, the dose in the TSC from DBAs is discussed in DCD Tier 2, Subsection 15.6.5.5.1.3 as being represented by the MCR consequences. The staff's review of the TSC habitability is discussed in further detail below in Section 15.0.3.4.11 of this SER.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the LRA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the LRA. Therefore, the staff has determined that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis LRA meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.6 Radiological Consequences of Rod Ejection Accident (REA)

The applicant submitted a radiological consequence analysis for the REA in DCD Tier 2, Section 15.4.8.5. The mechanical failure of a control rod mechanism pressure housing is postulated to result in the ejection of a rod cluster control assembly (RCCA) and drive shaft. Because of the resultant opening in the pressure vessel, primary coolant is lost to the containment with concurrent rapid depressurization of the reactor pressure vessel. This mechanical failure causes a rapid positive reactivity insertion, together with an adverse core power distribution, possibly leading to localized fuel rod damage. The analysis assumes a LOOP after the reactor trip.

In accordance with the guidance in RG 1.183 the applicant evaluated the following two release cases:

- Primary containment leakage pathway, assuming that the entire activity released from the fuel becomes airborne in the primary containment and available for release through containment leakage.

- Secondary side leakage pathway, assuming that the entire activity released from the fuel is retained in the RCS and is available for release through SG tube leakage from the primary coolant to the secondary coolant and secondary side steaming.

The applicant added the consequences of the two cases together, which is conservative compared to the guidance in RG 1.183, where the cases are evaluated separately.

The applicant assumed that 10 percent of the fuel rods experience DNB, with release of all the activity in the fuel-cladding gap to the primary coolant. In addition, 0.25 percent of the fuel in the core is assumed to melt. Although the applicant assumed some fuel melting, the applicant states that the US-APWR design ensures that the post-REA peak fuel temperature will remain below incipient fuel melting conditions, consistent with the guidance in SRP section 4.2, Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents." The staff's review of the fuel system design is discussed in section 4.2 of this SER.

The radioactivity release from fuel clad failure assumed that the fuel rods have been operating at a radial peaking factor of 1.78. Ten percent of the fuel rod noble gas and halogen inventory

and 12 percent of the alkali metal inventory, are assumed to be in the fuel-cladding gap and released initially, which is consistent with the guidance in RG 1.183. The transient causes an additional fission product release of 11 percent of the fission products from the failed fuel rods. This results in a total release of 21 percent of the fission product inventory from the fraction of the core with failed cladding. The applicant's assumptions on radioactivity release from fuel overheat and melting is consistent with the guidance in RG 1.183, Appendix H.

In RAI 38-412, Question 15.00.03-6, the staff requested that the applicant provide the basis for assuming 0.25 percent of the core fuel melts as a result of the REA. In its August 22, 2008, letter, the applicant responded that the 0.25 percent fuel melting assumption is conservative, considering that analyses predict no fuel centerline melting for the rod ejection accident. The value of 0.25 percent corresponds to a case where fuel melting occurs in 10 percent of the fuel rods in DNB, which is less than 10 percent of the rods in the core, and the melting around the fuel centerline is assumed to be one-quarter of the fuel rod volume. Considering that analyses show that no fuel melting is predicted for the rod ejection accident, the staff finds the assumption of 0.25 percent fuel melting in the radiological consequence analysis to be conservative and bounding, and **RAI 38-412, Question 15.00.03-6, is resolved**.

In RAI 38-412, Questions 15.00.03-7 and -8, the staff requested that the applicant provide additional information on the transient fission product release for the REA. In its August 22, 2008, response, the applicant stated that the modeling of the transient fission product release is based on guidance provided in SRP Section 4.2, Appendix B, and provided a discussion of the calculation. The staff finds the applicant's calculation conforms to the guidance in SRP Section 4.2, Appendix B, and is therefore acceptable. The staff considers **RAI 38-412, Questions 15.00.03-7 and 8, resolved**.

The applicant assumed that the release of fission products to the environment may occur via either of two pathways. The containment leakage first pathway involves a release of primary coolant to the containment, which is assumed to leak into the environment at the design leak rate of the containment. The applicant's analysis took credit for aerosol and iodine removal by spray and natural deposition in the containment, using the same assumptions as in the LOCA analysis. The staff's review of the containment aerosol deposition and iodine removal is discussed below as part of the review of the LOCA analysis in Section 15.0.3.4.10 of this SER.

In the secondary-side leakage pathway, fission products would reach the secondary coolant via the SGs with a maximum total allowable primary-to-secondary leak 0.394 l/min (150 gpd) of primary-to-secondary leakage through any SG, as specified in the US-APWR TS. For both pathways, the applicant assumed that the US-APWR reactor operated at its TS equilibrium value of 37 kBq/gm (1 μCi/gm) for DEI-131 and 11.1 MBq/gm (300 μCi/gm) for DE Xe-133.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the REA. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the REA.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis REA is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated rod ejection accident will fall well within the dose criterion set forth in 10 CFR 50.47 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

The applicant stated that the doses in the MCR and TSC for the REA are bounded by the doses calculated for the LOCA, and provided dose estimates for the MCR. Modeling of the TSC is not discussed in detail in DCD Tier 2, Section 15.4.8.5, nor are TSC dose results for the REA provided. However, the dose in the TSC from DBAs is discussed in DCD Tier 2, Subsection 15.6.5.5.1.3 as being represented by the MCR consequences. The staff's review of the TSC habitability is discussed in further detail below in Section 15.0.3.4.11 of this SER.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the REA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the REA. Therefore, the staff has determined that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis REA meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.7 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," includes a provision to ensure isolation of all pipes that are part of the RCPB and penetrate the containment building. GDC 55 also provides that small-diameter pipes that must be continuously connected to the primary coolant system to perform necessary functions may be acceptable based on some other defined bases. For these lines, methods of mitigating the consequences of a rupture are necessary because the lines cannot be isolated automatically. For the US-APWR design, the applicant determined that the bounding small lines in this category are the reactor coolant sample lines and CVCS letdown line to the demineralizers. The applicant submitted a radiological analysis for a small-line failure in DCD Tier 2, Section 15.6.2.5.

The CVCS line break is downstream of the heat exchanger, which reduces the temperature, which leads to no coolant flashing at the CVCS break. Therefore iodine release to the environment is relatively insignificant. The release from the CVCS letdown line break outside containment is bounded by the reactor coolant sample line break, and dose results for the CVCS line break are not reported in the DCD. The flow from the sample line is passively restricted by the size of the line itself, and the release ends when isolation by manual action is assumed to occur at 45 minutes after the break. The break flow rate at the break point was calculated using the RCS pressure, temperature, and break size. The flow resistance of the sample line piping and valves was conservatively neglected. The calculated flow rate was adjusted for the assumed density of 1 gm/cubic centimeter (cc) (62.4 lb/ft^3). The staff finds the calculation of the break flow rate is representative of the expected accident conditions and the assumed break flow rate is acceptable.

For the small-line break, fuel damage is not assumed because the loss of primary coolant is relatively small and is compensated by the automatic makeup system. The primary coolant activity concentrations are assumed to be initially at the proposed TS RCS equilibrium limit of 37 kBq/gm (1 µCi/gm) for DEI-131 and 11.1 MBq/gm (300 µCi/gm) for DE Xe-133. The applicant assumed an accident-initiated iodine spike in the primary coolant is caused by the postulated reactor shutdown or depressurization. The iodine spike raises the equilibrium iodine appearance rate by a factor of 500, in accordance with the guidance in SRP 15.6.2.

The fraction of the iodine in the released coolant that becomes airborne and available for release to the atmosphere is assumed to be equal to the fraction of the coolant that flashes to steam. Based on the thermodynamic conditions and enthalpy, 47 percent of the leaked reactor coolant is calculated to flash to steam. Noble gases released from the RCS are assumed to be released directly to the environment without mitigation. The staff finds the assumptions on the amount of released coolant fission products that are assumed to be airborne acceptable and in agreement with guidance on assumptions for radioactivity released from a small line break found in SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment."

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment are consistent with guidance in RG 1.183 on similar events and SRP Section 15.6.2, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the small-line break accident. The staff's analyses followed guidance in SRP Section 15.6.2 and RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the small-line break.

Based on the comparison of the applicant's analysis methodology to the regulatory guidance and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis small-line break is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated small-line break will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.47.

The applicant stated that the doses in the MCR and TSC for the small-line break are bounded by the doses calculated for the LOCA, and provided dose estimates for the MCR. Modeling of the TSC is not discussed in detail in DCD Tier 2, Section 15.6.2.5, nor are TSC dose results for the small-line break provided. However, the dose in the TSC from DBAs is discussed in DCD Tier 2, Subsection 15.6.5.5.1.3 as being represented by the MCR consequences. The staff's review of the TSC habitability is discussed in further detail below in Section 15.0.3.4.11 of this SER.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the small line break, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR

and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the small line break. Therefore, the staff has determined that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis small line break accident meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.8 Radiological Consequences of Steam Generator Tube Rupture (SGTR)

The applicant has evaluated the radiological consequences of a postulated SGTR accident and provided a radiological consequence analysis for the accident in DCD Tier 2, Section 15.6.3.5. This DBA assumes that a single tube in one SG fails, releasing primary coolant to the secondary side of the affected SG. The analysis assumes a LOOP after the reactor trip and the MSRV of the affected SG fails in the fully open position. Adequate core cooling precludes fuel failure. Following the guidance in RG 1.183, the applicant considered two coolant activity concentration cases:

For the accident-induced iodine spiking case, the analysis assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the SGTR. Before the postulated accident, the US-APWR reactor was assumed to operate at the TS 3.4.16 equilibrium iodine concentration limit of 37 kBq/gm (1.0 µCi/gm) for DEI-131 and 11.1 MBq/gm (300 µCi/gm) for DE Xe-133 in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 335, resulting in a rising iodine concentration in the primary coolant during the course of the accident.

For the pre-accident iodine spiking case, the analysis assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum instantaneous concentration limit of 2.2 MBq/gm (60 µCi/gm) for DEI-131, specified in the US-APWR TS.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183 guidance, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the two coolant activity scenarios for the SGTR accident. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis SGTR is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated SGTR accident with accident-induced iodine spiking will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.47.

The staff also concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the

radiological consequences of a postulated SGTR accident with pre-accident iodine spiking in the coolant will not exceed the dose criterion set forth in 10 CFR 50.47 (i.e., 0.25 Sv (25 rem) TEDE).

The applicant stated that the doses in the MCR and TSC for the SGTR are bounded by the doses calculated for the LOCA, and provided dose estimates for the MCR. Modeling of the TSC is not discussed in detail in DCD Tier 2, Section 15.6.3.5, nor are TSC dose results for the SGTR provided. However, the dose in the TSC from DBAs is discussed in DCD Tier 2, Subsection 15.6.5.5.1.3 as being represented by the MCR consequences. The staff's review of the TSC habitability is discussed in further detail below in Section 15.0.3.4.11 of this SER.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the SGTR, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for each of the SGTR iodine spiking cases. Therefore, the staff has determined that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis SGTR meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.9 Radiological Consequences of Fuel Handling Accident and Cask Drop

In DCD Tier 2, Section 15.7.4, the applicant presented its analysis of the radiological consequences of a postulated fuel-handling accident (FHA). For the US-APWR design, an FHA can be postulated to occur either inside the containment or in the fuel handling area. The analysis assumptions are bounding for the FHA in either location. The applicant assumed, in accordance with guidance in RG 1.183, that fission products are released directly to the environment within a 2-hour period without credit for any iodine removal processes except for retention in overlying fuel pool water.

Other FHAs such as a spent fuel cask falling or tipping onto the spent fuel pool are prevented by the design of the spent fuel handling equipment. DCD, Tier 2, Section 15.7.5, states that the spent fuel cask and transfer machine are located so that a cask is prevented from being in the spent fuel pool area altogether. The overhead heavy load system is designed with a single-failure-proof crane, precluding the need to perform heavy load drop evaluations. Therefore, the applicant did not provide an analysis of the cask drop event. No accident analysis for a spent fuel cask drop is necessary. Therefore, the applicant's discussion of the cask drop is acceptable.

For the FHA, the applicant assumed that a single fuel assembly that has undergone 24 hours of decay time is dropped, such that the activity in the gap of every rod in the dropped assembly is released. The kinetic energy of the falling fuel assembly is assumed to break open the maximum possible number of fuel rods. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (8 percent of I-131, 10 percent of Kr-85, and 5 percent of other iodine and noble gas inventories in the fuel rod) is assumed to occur, with the released gases bubbling up through the fuel pool water. These gap fractions conform to RG 1.183 guidance, and are acceptable for use in the FHA analysis because the fuel burnup and linear heat generation rate limitations in the footnote to RG 1.183 Table 3 are met for the US-APWR

fuel. The applicant assumed an effective decontamination factor of 200 for total iodine as it rises through the fuel pool water. The fuel pool water depth above the fuel is at minimum 7.01 m (23 ft); therefore, in accordance with the guidance in RG 1.183, the decontamination factor of 200 is acceptable. The applicant assumed that iodine in the particulate form is not volatile and, therefore, is not released. In accordance with RG 1.183 guidance, the applicant assumed that the particulate CsI is converted instantaneously to the elemental form of iodine when it is released from the fuel into the low-pH pool water. Although the containment might be isolated when movement of fuel is taking place, the applicant's analysis assumed that the release from the pool goes directly to the environment without holdup or mitigation except for retention in overlying fuel pool water.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the FHA. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the FHA.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis FHA is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated FHA will fall well within the dose criterion set forth in 10 CFR 50.47 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

The applicant stated that the doses in the MCR and TSC for the FHA are bounded by the doses calculated for the LOCA, and provided dose estimates for the MCR. Modeling of the TSC is not discussed in detail in DCD Tier 2, 15.7.4, nor are TSC dose results for the FHA provided. However, the dose in the TSC from DBAs is discussed in DCD Tier 2, 15.6.5.5.1.3 as being represented by the MCR consequences. The staff's review of the TSC habitability is discussed in further detail below in Section 15.0.3.4.11 of this SER.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the FHA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the FHA. Therefore, the staff has determined that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis FHA meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.10 Radiological Consequences of LOCAs

In DCD Tier 2, Section 15.6.5.5, the applicant provided the analysis of a hypothetical design-basis LOCA for radiological consequences. The applicant concluded that certain bounding sets of atmospheric relative concentration values (χ/Qs), specified in DCD Tier 2,

Section 2.3 as site parameters, in conjunction with the use of containment sprays, natural deposition of fission product aerosol within the containment, and the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design-basis LOCA will fall within the relevant dose criteria established in 10 CFR 52.47 and in GDC 19.

The design-basis LOCA analyzed for radiological consequences is a postulated accident that results from primary coolant loss in excess of the RCS makeup capacity, thereby leading to meltdown of all assemblies in the core. The DBA LOCA is analyzed to determine whether the primary containment is sufficiently capable in preventing release of fission products to the outside environment and also determine whether the fission product mitigation systems and features are sufficient as compared to the referenced regulatory dose criteria. The applicant followed the guidance in RG 1.183 in performing the LOCA radiological consequences analysis.

The applicant used the core radionuclide inventory discussed above in 15.0.3.1 of this report, which assumes operation at 102 percent of rated thermal power or 4540 MWt. The applicant also assumed that reactor coolant radionuclide concentrations were at the TS equilibrium limits for noble gases and iodines, and the particulate concentration is based on 1-percent fuel defect, consistent with the coolant source terms in DCD Section 11.1. Consistent with the guidance in RG 1.183, the analysis considered releases through three pathways; containment purge release prior to purge system isolation, primary containment leakage, and ESF component leakage. The LOCA analysis assumed a coincident LOOP.

<u>Containment Leakage Pathway</u>

All releases from the core and RCS are assumed to mix instantaneously and homogenously in the primary containment atmosphere as they are released. At the onset of the LOCA, the containment purge system is assumed to be in operation, and the purge flow continues until the primary containment is isolated at 15 seconds. No credit was taken for filtration of the purge exhaust. After isolation, release of the containment atmosphere to the outside environment is through containment leakage at the proposed TS containment leakage rate limit of 0.15 percent per day for the first 24 hours and half that value afterward, until the end of the accident at 30 days. No holdup was credited in the secondary buildings. Primary containment leakage to the penetration areas and safeguard component areas is capable of being filtered before release to the environment during the accident, after those areas are brought to a negative pressure with respect to the adjacent areas. During the annulus emergency exhaust system pressure drawdown period of 4 minutes after onset of the accident, 100 percent of the primary containment leakage was assumed to be released directly to the environment as a ground-level release. Following the completion of the annulus drawdown, the primary containment leakage to the penetration areas (i.e., 50 percent of the total primary containment leakage) was assumed to be filtered by high-efficiency particulate air (HEPA) filters at 99-percent efficiency for particulates and released through the plant stack as a ground-level release. After annulus drawdown, the remaining 50 percent of the primary containment leakage was assumed to be released directly to the environment as a ground-level release. The staff’s review of the containment and annulus design and testing is discussed in Section 6.5 of this SER.

The applicant’s analysis takes credit for aerosol natural deposition in the containment based on the model described in NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments,” incorporated into RADTRAD as the Powers model for

containment aerosol natural deposition. The applicant used the 10^{th-}percentile removal coefficients in the Powers natural deposition model, in accordance with the DBA analysis guidance in RG 1.183.

In RAI 38-412, Question 15.00.03-23, the staff noted that the Powers natural deposition model is correlated to reactor type using operating PWR and boiling water reactor (BWR) information on containment geometry and power, and requested the applicant explain why the Powers natural deposition correlation is applicable to the US-APWR containment. In its August 22, 2008, response, MHI provided a discussion comparing the US-APWR containment volume and power to the information used as a basis for developing the Power aerosol natural deposition correlation given in NUREG/CR-6189. The Powers aerosol natural deposition coefficients are correlated to thermal power, while assuming that the containment volume is also correlated to the thermal power, within the bounds of plants used as a basis for the correlation. The applicant's discussion shows that the US-APWR containment volume falls within the bounds assumed for a plant with nominal power level of 4451 MWt. The staff finds that MHI's discussion shows that the use of the Powers aerosol natural deposition correlation is acceptable for the US-APWR. Based on the applicant's response, the staff considers RAI 38-412, Question 15.00.03-23 resolved.

The applicant also modeled aerosol removal by the containment spray system (CSS). The CSS is an ESF system intended for use in mitigating a design basis accident for heat and fission product removal. The CSS consists of four 50-percent-capacity trains, of which only two are assumed to be operating for the DBA LOCA dose analysis. Review of the CSS design and function is discussed in Sections 6.2.2 and 6.5.2 of this SER. Aerosol removal by the CSS is assumed to occur in 60 percent of the containment volume, which is the percentage of the containment free volume that is sprayed. In accordance with the guidance in SRP Section 6.5.2, the applicant modeled the natural convection flow between the sprayed and unsprayed regions of the containment volume by the conservative assumption of flow between the regions equivalent to two times the volume of the unsprayed region per hour. The CSS initiates operation 5 minutes after the onset of the LOCA and operates continuously throughout the duration of the accident. The applicant's analysis used spray removal coefficients calculated using the method in SRP Section 6.5.2 for aerosol removal by sprays. In accordance with the guidance in SRP Section 6.5.2, the aerosol spray removal coefficient was decreased by a factor of 10 after the decontamination factor reached a value of 50 at 3.23 hours.

Removal of elemental iodine in the containment by wall deposition was modeled based on the guidance in SRP Section 6.5.2, and was assumed to end at about 15 hours when a total decontamination factor of 200 was reached. Organic iodine is not depleted by these processes.

ESF component leakage pathway

ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. The applicant's analysis used the RG 1.183, Appendix A guidance on the source term assumptions for the ESF system leakage, which states that with the exception of the noble gases, the same source term that was released to the containment in the containment leakage pathway should be assumed to be instantaneously and homogeneously mixed in the sump water as it is released from the fuel.

The applicant's analysis assumed that the ESF component total leakage is 8.0 kg/hr (17.6 lb/hr), which is twice the proposed limit, in accordance with the guidance in RG 1.183. A

10-percent flashing fraction was applied to model the amount of iodine in the leaked fluid that becomes airborne. The iodine available for release to the environment is assumed to be 97% elemental and 3% organic. Following the completion of the annulus emergency exhaust system drawdown, the ESF component leakage release is filtered at 99 percent efficiency for particulates and released through the plant stack as a ground-level release.

The ESF leakage rate was conservatively estimated assuming that portions of the Safety Injection System (SIS) and CSS that circulate water outside the containment leak during their intended operation and that the pumps, heat exchangers, instruments and valves in these systems leak. The ESF leakage rate is the sum of design leakage for the various system components. Guidance in RG 1.183, Appendix A, states that the ESF leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the TS or licensee commitments to item III.D.1.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” would require declaring such systems inoperable. The US-APWR TS do not include requirements on ESF leakage; however, the US-APWR TS Program 5.5.2, “Primary Coolant Sources Outside Containment,” does include requirements to minimize leakage in these systems by performing maintenance and visual inspection on a periodic basis and requiring integrated leak tests for each system at least once per 24 months. The staff finds that the applicant’s basis for the ESF leakage assumption as the sum of the design leakages, as supported by the TS 5.5.2 program, will sufficiently assure that the ESF leakage value used on the DBA LOCA radiological consequences analysis is appropriately representative of the post-accident ESF system leakage.

The staff has reviewed the applicant’s analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183 guidance, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant’s assessment, the staff performed independent radiological consequence calculations for the LOCA. The staff’s analyses followed guidance in RG 1.183 and used the applicant’s assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the LOCA.

The staff performed an independent MCR and TSC radiological consequence calculation for the LOCA, using the applicant’s assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The MCR and TSC radiological consequences calculated by the staff are consistent with those calculated by the applicant and confirm that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE.

Based on the comparison of the applicant’s analysis methodology to the guidance in RG 1.183 and the staff’s confirmatory analysis, the staff finds that the applicant’s analysis of the design basis LOCA is acceptable. The staff concludes that the US-APWR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated LOCA will not exceed the dose criterion set forth in 10 CFR 50.47 (i.e., 0.25 Sv (25 rem) TEDE). The staff also concludes that there is reasonable assurance that the radiological consequences in the MCR and TSC following a design basis LOCA meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.4.11 Main Control Room and Technical Support Center Radiological Habitability Analysis

DCD Tier 2, Subsection 15.6.5.5.12, discusses the radiological consequence analysis for personnel in the main control room during a design-basis LOCA, relying on the MCR emergency filtration system to limit the radioactivity to which the personnel may be exposed. The control room envelope includes the main control room and is served by the MCR heating, ventilation, and air conditioning (HVAC) system, as described in DCD Tier 2, Sections 6.4 and 9.4. The MCR emergency filtration systems are a part of the MCR HVAC system. Staff's review of the control room habitability and the control room ventilation systems is discussed in Section [6.4] and [9.4] of this report, respectively. The US-APWR TS include a control room envelope habitability program to maintain the systems and control room envelope.

The MCR HVAC system provides a slight positive pressure with the main control room area with respect to adjacent areas during normal and accident conditions. For an accident with release of radioactivity, the outside air supply is automatically diverted through the MCR emergency filtration system charcoal and HEPA filter trains. The system is actuated by an ECCS actuation signal or by high radiation levels in the air intake ducts. Filtered recirculation is initiated when the control room is isolated in emergency operation. Each DBA radiological consequence analysis conservatively modeled the main control room emergency ventilation system operation by assuming intake flow equivalent to operation of both of the two redundant charcoal filtration systems, but recirculation flow assuming the loss of one train of the two redundant charcoal filtration systems for the duration of the accident. A conservative assumption of an unfiltered inleakage rate of 3.40 m^3/min (120 cfm), including ingress/egress, is used in the DBA radiological consequences analyses. The applicant stated that the analysis assumed an unfiltered inleakage rate of 0.28 m^3/min (10 cfm) through ingress/egress through the control room vestibule doors. The staff finds this assumption is consistent with staff guidance in SRP 6.4 and is acceptable. The remainder of the assumed unfiltered inleakage is subject to the testing requirements in US-APWR TS Program 5.5.20, "Control Room Envelope Habitability Program."

In its control room dose analyses, the applicant adjusted the dose to the control room personnel due to external gamma radiation from airborne activity within the main control room by applying a finite-cloud correction, consistent with the guidance in RG 1.183.

The applicant's control room radiological consequences analyses also considered external dose from the following sources:

- Radiation shine from the external radioactive plume released from the facility
- Radiation shine from radioactive material in the containment
- Radiation shine from radioactive material in the MCR emergency filtration unit

The direct radiation shine doses are included in the calculated main control room TEDE for the DBAs. The applicant's calculation of the direct radiation shine dose used LOCA fission product sources, control room design dimensions, and selected the assumed receptor location for each source to maximize the shine dose to the operator. There is no specific guidance on performing direct dose analyses in RG 1.183. However, the staff has determined that the applicant has used direct dose analysis best practices to model the control room dimensions, source and receptor locations, and used acceptable radioactive material sources. Therefore, the staff finds that the MCR direct dose analysis is acceptable.

Although the TSC is not included in the control room envelope, the applicant stated that the control room radiological consequences analyses are representative for the TSC, based on the similarity of the design of TSC ventilation system and envelope to those for the control room. In RAI 38-412, Question 15.00.03-17, and RAI 105-1624, Question 15.00.03-25, the staff requested additional information on the TSC habitability analysis, including information on the atmospheric dispersion factor calculations, in order to assess the statement.

By letters dated August 22, 2008, and January 6, 2009, the applicant responded to RAI 38-412, Question 15.00.03-17, and RAI 105-1624, Question 15.00.03-25. The staff found the information to be sufficient to assess the DBA radiological consequences in the TSC, and confirmed that sufficient discussion of the TSC dose modeling, including atmospheric dispersion factors, was added to the DCD. Based on the applicant's response, the staff considers **RAI 38-412, Question 15.00.03-17, and RAI 105-1624, Question 15.00.03-25, resolved**.

In its independent dose analyses for each DBA, the staff calculated the radiological consequences in the TSC and compared the result to the calculated value for the MCR. In each case, the MCR dose bounded the TSC dose. Therefore, the staff was able to confirm the applicant's statement that the consequences of DBAs in the MCR are representative for the TSC, for the US-APWR design. The staff was also able to confirm that the radiological consequences of DBAs within the MCR and TSC meet the applicable regulatory requirements.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the TSC model and MCR model are reasonable and consistent with the guidance in SRP 6.4 and RG 1.183.

15.0.3.5 Combined License Information

COL Information Items from DCD Tier 2 Table 1.8-2

Table 15.0-1
US-APWR Combined License Information Items

Item No.	Description	Section(s)
COL 15.0(1)	*In the COLA, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and 10 CFR 52.79 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criterion 19 are met for affected events using site-specific χ/Q values. Additionally, the Technical Support Center (TSC) dose should be evaluated against the habitability requirements in Paragraph IV.E.8 to 10 CFR Part 50, Appendix E, and 10 CFR 50.47(b)(8) and (b)(11).*	15.0.3

15.0.3.6 Conclusions

The staff concludes that the information contained in the DCD Tier 2, Chapter 15 conforms to the guidance of RG 1.183 regarding control of the radioiodines in the post-LOCA environment as related to the iodine source term assumptions for use in DBA radiological consequences analyses.

The staff has reviewed the radiological consequences analyses of the DBAs described in DCD Tier 2, Chapter 15, for the US-APWR design. Based on the evaluation discussed above, the staff concludes that the US-APWR design meets 10 CFR 52.47(a)(2)(iv) dose criteria and the accident-specific offsite dose acceptance criteria, given in RG 1.183 and SRP Section 15.0.3.

The staff finds reasonable assurance that the main control room habitability systems, as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following DBAs to meet the dose criterion specified in GDC 19.

The staff finds reasonable assurance that the main control room habitability systems can mitigate the dose in the TSC following DBAs to be within 0.05 Sv (5 rem) TEDE, to meet the TSC habitability requirements in Paragraph IV.E.8 of Appendix E to 10 CFR Part 50, and 10 CFR 50.47(b)(8) and (b)(11).

15.0.3.7 References

1. T. S. Kress, E. C. Beahm, C. F. Weber, and G. W. Parker, “Fission Product Transport Behavior,” *Nucl. Technol*. **101**(3), 262 (1993).
2. E. C. Beahm, R. A. Lorentz, and C. F. Weber, *Iodine Evolution and pH Control*, NUREG/CR-5950 (ORNL/TM-12242), Oak Ridge National Laboratory (December 1992).
3. *Pressurized Water Reactor Primary Water Chemistry Guidelines*, Vol. 1, Rev. 6, Electric Power Research Institute (December 2007).
4. J. Wing, *Post-Accident Gas Generation from Radiolysis of Organic Materials*, NUREG-1081 (September 1984).
5. NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments,” USNRC, July 1996.
6. NUREG/CR-6604, “RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation,” USNRC, June 1997.
7. NUREG/CR-6604, Supplement 2 “RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation,” USNRC, October 2002.

15.1 Increase in Heat Removal by the Secondary System

15.1.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.1.1 Introduction

This section describes the evaluation of DCD Tier 2, Sections 15.1.1 “Decrease in Feedwater Temperature,” 15.1.2 “Increase in Feedwater Flow,” 15.1.3 “Increase in Steam Flow,” and 15.1.4 “Inadvertent Opening of a Steam Generator Relief or Safety Valve.” Each of these events could lead to an increase in the heat removal by the secondary system, which could result in a temperature decrease in the RCS. These events are discussed as a set below because they are all AOOs that abide by the same requirements and acceptance criteria.

15.1.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided DCD Tier 2 safety analyses in Section 15.1.1 through Section 15.1.4, summarized here as follows:

DCD Section 15.1.1 Decrease in Feedwater Temperature

A decrease in feedwater temperature is assumed to result from the functional loss of the high-pressure or low-pressure feedwater heaters. The decrease in feedwater temperature will increase heat transfer across the SGs and lower the temperature of the reactor coolant. With a negative moderator temperature coefficient, the positive reactivity insertion results in an increase in core power. For the limiting case, loss of a high-pressure feedwater heater at hot full power operation, the plant stabilizes at a new, higher power level. The RTS is not actuated and no systems are required to mitigate the event.

DCD Section 15.1.2 Increase in Feedwater Flow

An increase in feedwater flow is assumed to result from failure or misoperation of the main feedwater regulation valve during rated power or part load operation. The increase in feedwater flow to the affected SG causes a decrease in the reactor coolant temperature in the associated cold leg. With a negative moderator temperature coefficient, the positive reactivity insertion results in an increase in core power. For the limiting case of one full open main feedwater regulation valve at hot full power operation, the event is terminated when the high-high SG water level signal trips the reactor and isolates main feedwater.

DCD Section 15.1.3 Increase in Steam Flow

A rapid increase in steam flow can occur when the main steam flow is increased above the steady-state demand flow due to an error (administrative, operator or equipment malfunction) that causes the turbine bypass, main turbine control, main steam relief, or main steam depressurization valve to, inadvertently, fully open. The increase in steam flow causes a decrease in temperature at the reactor vessel inlet. With a negative moderator temperature coefficient, the positive reactivity insertion results in an increase in core power. For the limiting case at hot full power operation, the plant stabilizes at a new, higher power level. The RTS is not actuated and no systems are required to mitigate the event.

DCD Section 15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a main steam relief, main steam depressurization, main steam safety, or turbine bypass valve can cause depressurization of the secondary system, removing energy from the RCS and causing a reduction in reactor coolant temperature and pressure. With a negative moderator temperature coefficient, the positive reactivity insertion results in an increase in core power. From hot standby conditions, this event can lead to criticality and a brief return to low power until the low pressurizer pressure signal initiates the ECCS, which terminates the transient by injecting borated water into the core.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.1.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.1.1 – 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve," and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Sections 15.1.1-15.1.4.

1. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. 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.

4. GDC 20, as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs.

5. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by ensuring that the analysis accounts for appropriate margin for malfunctions such as stuck rods.

Acceptance criteria adequate to meet the above requirements include:

1. Identify which of the moderate-frequency initiating events that result in increased heat removal are the most limiting.

2. Verify that, for the most limiting initiating events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

3. Pressure in the reactor coolant and main steam systems is maintained below 110 percent of the design values.

4. Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations.

5. An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

6. To meet the requirements of GDCs 10, 13, 15, 20, and 26, the positions of RG 1.105, “Instrument Spans and Setpoints,” are used with regard to their impact on the plant response to the type of transient addressed in NUREG-0800.

7. The most limiting plant systems single failure is assumed in the analysis and satisfies the positions of RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems.”

8. The analyses of transients caused by excessive heat removal are performed using an acceptable analytical model, and approved methodologies and computer codes. The values of the parameters used in the analytical model are suitably conservative.

15.1.1.4 Technical Evaluation

Methods, models, and analysis assumptions common to all four events are discussed first, followed by an evaluation of event specific items.

The staff confirmed that the (1) decrease in feedwater temperature, (2) increase in feedwater flow, (3) increase in steam flow, and (4) inadvertent opening of a SG relief or safety valve events are simulated using the computer code MARVEL-M and methods described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

For events that credit a reactor trip, in addition to the limiting single failure, the analysis assumes LOOP occurs 3 seconds after the reactor/turbine trip. The applicant states it is not necessary to run a separate LOOP case because the no-LOOP case is bounding with respect to the relevant acceptance criteria (DNBR, primary system pressure and secondary system pressure). The staff agrees as discussed in Section 15.0.0.4 of this SER.

The DCD did not include transient SG pressure plots for the first three events; the applicant had indicated that this was not a key parameter. The staff issued RAI 301-2324, Question 15.1-3 requesting these plots. The applicant responded on June 16, 2009, with transient plots demonstrating that the SG pressures declined from their initial values for Events (1) and (3). For Event (2), the SG pressures rose from their initial values but stabilized at a value significantly below the acceptance criteria. The staff finds the response acceptable and agrees that overpressurization of the SG secondary system is not a key parameter for the analysis of these three events.

DNBR calculations for the first three events use the RTDP and WRB-2 DNB correlation. As prescribed by the RTDP, nominal values are used to define the initial conditions for reactor power, reactor coolant average temperature, and RCS pressure. These three events were run using the maximum moderator density coefficient and minimum Doppler power coefficient. The

staff agrees this minimizes the calculated DNBR because it provides the greatest positive reactivity and maximum power increase.

Numerical results for the minimum DNBR, primary system pressure and secondary system pressure are included in the applicant's response to RAI 297-2287, Question 15.0.0-16, demonstrating that the acceptance limits are not exceeded for these four events. No fuel failures are predicted; therefore, the radiological consequences for these events are bounded by the radiological consequences for the Section 15.1.5 MSLB, discussed in Section 15.0.3.4 of this SER.

Evaluations for Specific Events

<u>(1) Decrease in Feedwater Temperature (DCD Section 15.1.1)</u>

The applicant identified the limiting reduction in feedwater temperature as 30.6°C [55°F], caused by the instantaneous functional loss of a high-pressure feedwater heater at hot full power (HFP). This value is within the range of limiting temperature reductions documented for this event in safety analyses for other PWRs. The staff agrees HFP conditions are more severe than a no-load case because the rate of heat removal by the secondary system is reduced as the load and feedwater flow rate decline. No mitigating RTS or ESF systems are credited in the analysis; therefore, it is not necessary to assume a single failure.

This event was run with manual rod control. The applicant states that the use of automatic rod control does not need to be evaluated for this transient because the sensitivity studies performed for Event (3) in DCD Section 15.1.3 demonstrate there is no difference in results for manual and automatic rod control. The staff reviewed the studies and found that, while the automatic rod control cases had slightly more limiting DNBRs than the manual rod control case, this is acceptable because Event (1) has sufficient margin to the DNBR limit.

<u>(2) Increase in Feedwater Flow (DCD Section 15.1.2)</u>

The applicant identified the limiting event to be the full-open failure of one main feedwater regulation valve at HFP, causing one SG to be supplied with main feedwater at 300 percent of the rated loop flow. The staff agrees the case should be initiated at HFP because the no-load case will be bounded by the uncontrolled RCCA withdrawal event in Section 15.4.1.

The analysis credits a reactor trip and feedwater isolation initiated by a high-high SG water level signal. The delay time assumed for the reactor trip was found to be consistent with Tables 15.0-4. The staff will determine if the delay time assumed for feedwater isolation is appropriate upon resolution of Open Item 15.00-2.

In the description of this event, the DCD states that the reactor will stabilize at a new, higher, power level. The staff thought this was inconsistent with the evaluation (which credits a reactor trip), and issued RAI 787-5882, Question 15.01.01-15.01.04-8 asking for clarification. In a response dated August 24, 2011, the applicant proposed deleting the sentence regarding stabilization at a higher power level. The staff agrees this will remove the source of confusion and will track incorporation of the proposed change as **Confirmatory Item 15.01-1.**

A single failure was assumed in one train of the RTS, but this has no impact on the safety analysis because any one of the remaining trains is adequate to provide the trip function.

The staff issued RAI 301-2324, Question 15.1-2 to ask if the feedwater isolation valves are safety-related and if the feedwater isolation function is affected by a LOOP. The applicant responded on June 16, 2009, that the isolation valves are safety-related and do not require AC power for operation. The staff is satisfied that the feedwater isolation function is protected against single failures and is not affected by LOOP.

This event was run with manual rod control. The applicant states that the use of automatic rod control does not need to be evaluated for this transient because the sensitivity studies performed for Event (3) in DCD Section 15.1.3 demonstrate there is no difference in results for manual and automatic rod control. The staff reviewed the studies and found that the automatic rod control cases (with either maximum or minimum feedback) had slightly more limiting DNBRs than the manual rod control case. Because Event (2) has a more limiting DNBR than Event (3), the staff issued RAI 811-5958, Question 15.01.01-15.01.04-9, asking why it is not necessary to run the Event (2) case with automatic rod control. In a response dated September 30, 2011, the applicant provided sensitivity studies for Event (2) demonstrating that the automatic rod control cases (with either maximum or minimum feedback) are bound by the DCD manual rod control case with respect to DNBR and RCS pressure. The staff is satisfied with this response because it clearly shows the limiting event is captured in the DCD.

<u>(3) Increase in Steam Flow (DCD Section 15.1.3)</u>

The applicant identified the limiting event as a 10 percent step load increase initiated at HFP. As described in the response to RAI 297-2287, Question 15.0.0-6, other modes of operation are either bounded by the DCD case or by the inadvertent opening of a SG relief or safety valve transient discussed in Section 15.1.4. No mitigating RTS or ESF systems are credited in the analysis; therefore, it is not necessary to assume a single failure.

The applicant ran the following four cases to assess the impact of rod control and moderator reactivity feedback:

1) Manual rod control, minimum moderator density reactivity coefficient
2) Manual rod control, maximum moderator density reactivity coefficient
3) Automatic rod control, minimum moderator density reactivity coefficient
4) Automatic rod control, maximum moderator density reactivity coefficient

The staff reviewed the studies and found that while the automatic rod control cases had slightly more limiting DNBRs than the manual rod control cases, they demonstrated sufficient margin to the acceptance criteria.

<u>(4) Inadvertent Opening of a Steam Generator Relief or Safety Valve (DCD Section 15.1.4)</u>

The applicant modeled this event by assuming a steam release equal to the largest single main steam relief, main steam depressurization, main steam safety, or turbine bypass valve and by locating the valve upstream of the main steam check valve. The staff agrees that this is a bounding assumption because it results in a non-uniform cooldown of the reactor coolant system that cannot be terminated by the closure of the main steam isolation valves. The applicant initiated the event from hot standby conditions. The staff agrees with the applicant’s statement that if this event was initiated from full power, it would either be bounded by the increase in steam flow event discussed in Section 15.1.3 of this SER, or the reactor would trip causing a turbine trip. If the turbine trips, the post-trip conditions would eventually approach the

hot-standby conditions. However, during the time required to reach hot-standby conditions the steam flow would have decayed, reducing the effect of the cooldown, and making it less limiting than if the event was initiated from hot-standby conditions.

For this event, the RCS pressures are below the applicable pressure range for WRB-2 DNBR correlations and thus the DNBR is calculated using the ANC and VIPRE-01M computer codes and the W-3 correlation per the methodology described in MUAP-07010-P, Section 5.4. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER. Because a transient DNBR plot was not included in the DCD, it was requested in RAI 301-2324, Question 15.1-4 and provided in the applicant's June 16, 2009, response. The staff is satisfied with the resulting DNBR plot because it shows considerable margin to the acceptance criteria.

The analysis credits the low pressurizer pressure signal to actuate ECCS; safety injection into the reactor vessel ends the transient due to negative reactivity addition from the effects of boron injection. To maximize the cooldown, the analysis assumes that the EFWS operates from time zero until it is isolated by the low main steam line pressure signal. Because the sequence of events from the DCD did not include when the safety injection pumps start or when the EFW is isolated, these times were requested in RAI 787-5882, Question 15.01.01-15.01.04-7. In a response dated August 24, 2011, the applicant proposed DCD changes to add these events to Table 15.1.4-1, "Time Sequence of Events for Inadvertent Opening of a Steam Generator Relief or Safety Valve." The staff finds this response acceptable; it provided the requested information and the time delays associated with these events are consistent with DCD Tables 15.0-4 and 15.0-5. The staff will track incorporation of the proposed change as **Confirmatory Item 15.01-2.**

The analysis assumes a single failure of one train of the ECCS, resulting in operation of only two of the four safety injection pumps. The staff agrees this assumption is limiting because the other mitigating system, EFW isolation, is protected against single failure as each SG has two separate EFW isolation valves controlled by separate ESFAS trains.

15.1.1.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.1.1.6 Conclusions

With the exception of the confirmatory items discussed in this section, the staff concludes that the analyses of Sections 15.1.1 "Decrease in Feedwater Temperature," 15.1.2 "Increase in Feedwater Flow," 15.1.3 "Increase in Steam Flow" and 15.1.4 "Inadvertent Opening of a Steam Generator Relief or Safety Valve" are acceptable and meet the requirements of GDCs 10, 13, 15, 20 and 26. This conclusion is based upon the following:

- The staff has determined that the applicant's analysis was performed using a mathematical model that was found acceptable as discussed in Section 15.0.2.4 of this SER. The parameters used as input to this model were reviewed and found to be suitably conservative and consistent with the plant design. In addition, the staff has determined that the positions of RG 1.53 as related to the single failure criterion and RG 1.105 for instruments have also been satisfied.
- The applicant has met the requirements of GDCs 10, 20, and 26 with respect to demonstrating that the resultant fuel integrity is maintained since the SAFDLs were not exceeded for this event.

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of automatic protection systems occurred at values of monitored parameters that were within the instruments' prescribed operating ranges. No manual protection systems are credited.

- The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by these events and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

- The applicant has met the requirements of GDCs 20 and 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the SAFDLs were not exceeded.

15.1.2 Increase in Feedwater Flow

Review of this section of the DCD is documented under Section 15.1.1 of this SER.

15.1.3 Increase in Steam Flow

Review of this section of the DCD is documented under Section 15.1.1 of this SER.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Review of this section of the DCD is documented under Section 15.1.1 of this SER.

15.1.5.1 Steam System Piping Failures Inside and Outside of Containment

15.1.5.2 Introduction

A steam system piping failure inside or outside of containment can result in an increase in the heat removal capability of the secondary system, resulting in an unplanned increase in core power.

15.1.5.3 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.1.5, summarized here as follows:

A postulated MSLB removes heat from the RCS, which lowers RCS temperatures and pressures. In the presence of a negative moderator temperature coefficient, the cooldown of the RCS results in an insertion of positive reactivity. If the event occurs at nominal operating conditions, the core power increases. If the event occurs at hot zero power, the core could become critical and return to power. The core is ultimately shut down by the injection of boron into the RCS, the depletion of SG inventory, or a combination of the two. The analysis considers bounding cases to envelop the various assumptions on break size, break location, core power level, offsite power condition, and single failure. Mitigating systems are the reactor trip system, main steam line isolation, MFW isolation, safety injection, and EFW isolation.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.1.5.4 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.1.5 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.1.5 of NUREG-0800.

1. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. 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. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

3. GDC 27 and GDC 28, as they relate to the RCS being designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

4. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

5. GDC 35, as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model, which includes the potential adverse effects of hydraulic instabilities that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. The radiological criteria used in the evaluation of steam system pipe break accidents appear in SRP Section 15.0.3.

4. The integrity of the RCPs should be maintained such that loss of ac power and containment isolation will not result in pump seal damage.

5. The auxiliary feedwater system or other means of decay heat removal must be safety related and, when required, automatically initiated.

6. Tripping of the RCPs should be consistent with the resolution to Task Action Plan Item II.K.3.5.

15.1.5.5 Technical Evaluation

The staff confirmed the analysis was performed using the computer codes MARVEL-M, ANC and VIPRE-01M in accordance with the methodology described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

The applicant selected the following cases to bound all steam system piping break sizes, break locations, core power level and offsite power availability:

- Case A: Double-ended break from hot standby with offsite power available
- Case B: Double-ended break from hot standby without offsite power
- Case C: Spectrum of breaks from power with offsite power available

The events initiated at hot standby assume the largest break flow area, 0.13 square meters (m^2) [1.4 square feet (ft^2)]. The effective break area is limited by the flow restrictor integral to the steam generator outlet nozzle in the US-APWR SG design. The staff agrees that using the largest break results in the largest return to power as well as the largest radial peaking factor. For Case C, a series of runs was made encompassing a range of break sizes and initial power levels. For small breaks, the reactor does not trip and the plant stabilizes at a new higher power. For intermediate breaks, the power increase causes an overpower ΔT reactor trip. For large breaks (including those initiated from hot standby) a low steam line pressure signal (lead/lagged) actuates ECCS, which in turn trips the reactor, starts safety injection pumps, isolates main steam line and feedwater and actuates EFW. The low steam line pressure signal (not lead/lagged) also causes isolation of EFW to the faulted SG.

The calculations for the intermediate and large breaks from power are terminated shortly after the reactor trip because the core response after this time is bounded by the hot-standby cases. The staff agrees with this approach because during the time it takes for the event from power to reach the hot-standby conditions, the steam flow will have decayed, reducing the effect of the

cooldown. The response to RAI 297-2287, Question 15.0.0-6, explains why the DCD cases bound all operating modes, and the staff concurs with this assessment.

In the sequence of events for the large breaks, all four SGs blow down until the MSIVs close, after which one SG continues to blow down. This approach bounds all main steam system break locations because it models both a steam piping failure upstream of the check valve (with an assumed failure of the affected check valve to account for the initial steam release from the other SGs) and a steam piping failure downstream of the check valve (with an assumed failure of one isolation valve to account for the single SG blowdown after isolation). The staff agrees this represents a bounding case and notes that the valve failures are assumed in addition to the limiting single failure discussed later in this section.

ECCS actuation results in an automatic RCP trip. Therefore, the RCPs do not experience low-pressure or high-temperature fluid conditions that could lead to cavitation and potential damage during MSLB events, assuring the integrity and subsequent operability of the RCPs.

<u>Cases A and B</u>

The hot-standby cases are erroneously referred to as initiated from “hot zero power” and “hot shutdown” conditions in several places in the DCD. The analyzed cases are for hot standby, operating Mode 3, at no-load conditions with the reactor subcritical and with control rods inserted (except for the single most-reactive rod assembly, which is assumed to remain fully withdrawn). Hot zero power is a special case of startup, operating Mode 2, also at no-load conditions, but with the rods withdrawn and with the reactor critical. Hot shutdown is operating Mode 4, at low temperatures with the reactor subcritical and with control rods inserted. The terminology confusion is due to the fact that the MARVEL-M model must be initialized with zero reactivity, and in the analysis process the model is initialized at zero power with the control rods withdrawn. The control rods are then inserted rapidly at the beginning of the calculation to simulate the shutdown reactivity associated with hot-standby Mode 3 operation.

For the hot-standby cases, the limiting single failure is one train of the ECCS, either directly or as a consequence of an emergency gas turbine generator (GTG) failure. Under this assumption, the operation of only two of the four safety injection systems (SISs) is credited (one is assumed to fail and one is assumed to be out of service for maintenance). The staff concurs and notes that single failures in the other mitigating systems (reactor trip, main steam line isolation, feedwater isolation and EFW isolation) do not result in loss of function.

The hot-standby cases are run with and without offsite power available to account for the additional time delays associated with the startup and loading of the emergency GTGs. For Case A (offsite power available), the analysis conservatively does not credit the RCP trip that occurs on an ECCS signal. The staff agrees that assuming forced reactor coolant flow maximizes the RCS cooldown and the subsequent return to power. For Case B (LOOP) even though there is no turbine trip or subsequent grid disturbance, the analysis assumes power is lost at 4.5 seconds (the time of the ECCS actuation signal). This is 3 seconds after the time at which the steam line pressure reaches the low-pressure analytical limit, which would trip the reactor and turbine from a hot zero power initial condition. The staff issued RAI 302-2327, Question 15.1.5-1, requesting the basis for assuming RCP trip at that time. In a July 3, 2009, response, the applicant stated that ECCS actuation results in automatic RCP trip. Therefore, delaying RCP trip until the time of the ECCS actuation signal maximizes the period of forced reactor coolant loop flow, which is conservative. The staff finds the response acceptable because it maximizes the power increase.

The analysis credits the low main steam line pressure signal to actuate ECCS, isolate main steam lines and isolate EFW to the faulted SG. The time delays associated with these events are consistent with DCD Tables 15.0-4 and 15.0-5 (upon incorporation of **Confirmatory Item 15.00-4**).

For Cases A and B, the RCS pressures are below the applicable pressure range for the WRB-2 DNBR correlation; thus the minimum DNBR is calculated using ANC and VIPRE-01M and the W-3 correlation. Because DNBR plots were not included in the DCD, they were requested in RAI 302-2327, Question 15.1.5-4. In a July 3, 2009, response, the applicant provided the DNBR plots. The staff notes that while the DNBR for offsite power available Case A is more limiting than Case B, both results have margin and are also bounded by Case C.

The SRP Section 15.1.5 acceptance criteria on initial plant conditions state that the value assumed for the initial core flow should be justified because it is not clear which extreme (minimum or maximum) is more conservative. The staff asked for justification of the assumed minimum core flow in RAI 788-5883, Question 15.01.05-7. In a response dated August 24, 2011, the applicant provided sensitivity studies based on the limiting hot-standby event, Case A. The staff agrees that only hot-standby cases need to be considered because transients initiated from power (Case C) use the RTDP, which is designed to address uncertainties internally. The sensitivity studies showed that the case with maximum core flow produced a slightly higher core average heat flux and return to power than the DCD case. However, the applicant stated that this is one of the DCD cases that conservatively combine the maximum negative Doppler temperature coefficient with the minimum Doppler power coefficient (discussed in Section 15.0.0.4 of this SER). When this conservatism is removed (both Doppler coefficients set to the minimum value), the core average heat flux is less than the DCD case. The staff finds this response acceptable because it demonstrates the combination of parameters assumed in the DCD is suitably conservative for Cases A and B (which have less limiting DNBRs than Case C).

Case A was one of the six sample transient events included in MUAP-07010-P and thus is thoroughly reviewed as part of the MUAP-07010-P approval process. Case A was also one of the events selected for confirmatory calculations performed by the staff using RELAP5/MOD3.3. The results of the confirmatory calculations show good agreement with the DCD analysis. The main difference is that RELAP5/MOD3.3 calculated a slightly slower decrease in RCS pressure, which delayed the onset of ECCS injection flow, resulting in a slightly higher core power response.

<u>Case C</u>

For the cases initiated with the reactor at power, the positive reactivity insertion from the RCS cooldown immediately leads to increasing core power and (for intermediate and large breaks) a reactor trip. The analysis calculations are terminated shortly after the time of reactor trip because the results subsequent to trip are bounded by Cases A and B.

The intermediate and large break events credit a reactor trip; therefore, in addition to the limiting single failure, the analysis assumes LOOP occurs 3 seconds after the reactor/turbine trip. The applicant states it is not necessary to run a separate LOOP case because the no-LOOP case is bounding with respect to the relevant acceptance criteria (DNBR, primary system pressure and secondary system pressure). The staff agrees, as discussed in Section 15.0.0.4 of this SER.

For the Case C analyses, the reactor trip results in significant negative reactivity addition due to insertion of the rods (the single most-reactive rod assembly is assumed to remain fully withdrawn). The analysis does not credit the negative reactivity insertion from ECCS injection (because the transient ends prior to pump startup); the only safety system credited is the RTS. A single failure was assumed in one of the four RTS divisions, but this does not result in loss of the reactor trip function and thus there is no impact of the single failure on the safety analysis results.

The DNBR calculations for Case C use the RTDP and WRB-2 DNB correlation. As prescribed by the RTDP, nominal values are used to define the initial conditions for reactor power, reactor coolant average temperature, and RCS pressure. These transient cases use the maximum moderator density feedback which provides the greatest positive reactivity insertion and the maximum power increase.

From DCD Figure 15.1.5-26, "Initial Steam Flow, Peak Power, and Minimum DNBR versus Break Area, Steam System Piping Failure – Case C: Spectrum of Breaks from Power Conditions with Offsite Power," it appears the DNBR limiting event for Case C is an approximately .04 m^2 [0.4 ft^2] break initiated at 100 percent power. Because this specific event is more limiting than either Case A or Case B, the staff issued RAI 788-5883, Question 15.01.05-6, requesting the applicant provide the sequence of events identifying mitigating system actuations and time-related variations of key parameters in order to demonstrate that the acceptance criteria of SRP Section 15.1.5 are met. In a response dated August 24, 2011, the applicant proposed adding to the DCD a description of the limiting Case C event, a table of the sequence of events, and plots of key parameters. This analysis was based on 10 percent SG tube plugging and, as discussed in Section 15.0.0.4 of this SER, the staff does not agree this is a suitably conservative assumption for Case C because of the relatively low margin to the DNBR limit. The staff issued follow-up RAI 865-6151, Question 15.01.05-8 requesting that the DCD incorporate the more limiting assumption of 0 percent SG tube plugging in order to demonstrate the selected analysis parameters cover the predicted operating range. The staff also asked for a plot of SG pressure versus time to demonstrate the acceptance criteria regarding secondary pressure were met. The applicant responded on December 20, 2011, with a proposal to revise the DCD changes (originally included in response to Question 15.01.05-6) to utilize the 0 percent SG tube plugging assumption and to add a plot of transient SG pressure. The staff will track inclusion of these proposed changes in the next revision of the DCD as **Confirmatory Item 15.01-3**. In addition, the applicant revised the responses to RAI 297-2287, Questions 15.0.0-10 (summary of input parameters) and 15.0.0-16 (numerical results for acceptance criteria) in order to incorporate the new Case C analysis. The response to Question 15.0.0-16 demonstrates that the DNBR, peak pressure and secondary system pressure meet the AOO acceptance limits and no fuel failures are predicted. The radiological consequences of a MSLB are evaluated in Section 15.0.3.4.4 of this SER.

15.1.5.6 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.1.5.7 Conclusions

With the exception of the confirmatory items discussed in this section, the staff concludes that the consequences of postulated steam line breaks meet the relevant requirements set forth in GDCs 13, 17, 27, 28, 31, and 35 regarding (1) the ability to insert the control rods and to cool the core and (2) TMI Action Plan items. This conclusion is based upon the following:

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available and that automatic actuations of protection systems occurred at values of monitored parameters that were within the instruments' prescribed operating ranges. No credit is taken for operator actions.

- The applicant has met the requirements of GDCs 27 and 28 by demonstrating that for all cases the minimum DNBR experienced by any fuel rod remains above the 95/95 limit and no fuel failures are predicted.

- The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the PA. The maximum pressure remains below 110 percent of the design values.

- The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection).

- The staff finds the analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power (as required by GDC 17), were evaluated using a mathematical model that was found acceptable by the staff, as discussed in Section 15.0.2 of this SER.

- The parameters used as input to this model were reviewed and found to be suitably conservative.

- The radioactivity release is discussed in 15.0.3.4 of this SER.

- The applicant has met the requirements of 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) with respect to demonstrating the adequacy of the design of auxiliary feedwater or other qualified systems to remove decay heat following steam system piping failures.

- The applicant has met the requirements of 10 CFR 50.34(f)(1)(iii) with respect to demonstrating the integrity and operation of the RCPs to withstand the PA.

15.2 Decrease in Heat Removal by the Secondary System

15.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve, and Steam Pressure Regulator Failure

15.2.1.1 Introduction

This section documents the staff's review of DCD Tier 2, Sections 15.2.1 "Loss of External Load," 15.2.2 "Turbine Trip," 15.2.3 "Loss of Condenser Vacuum" and 15.2.4 "Closure of Main Steam Isolation Valve." Each of these events could result in a decrease in the rate of heat removal by the secondary system, which in turn, could lead to a temperature increase in the RCS and a pressure increase in both the RCS and the SG secondary side. These events are discussed as a set below because they are AOOs that abide by the same requirements and acceptance criteria.

15.2.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided DCD Tier 2 system descriptions in Section 15.2.1 through Section 15.2.4, summarized here as follows:

DCD Section 15.2.1 Loss of External Load

The LOEL is modeled by assuming an instantaneous step load decrease in both steam flow and feedwater flow from their full value to zero at the beginning of the transient. The sudden reduction in steam flow leads to an increase in pressure and temperature in the shell side of the SGs. As a result, the reactor coolant system temperature and pressure increase, the coolant density decreases, and the pressurizer water volume increases. Depending on the magnitude of the LOEL, the RTS, MSSVs, and pressurizer safety valves may be required to mitigate the transient.

DCD Section 15.2.2 Turbine Trip

In a turbine trip event, the main turbine stop valves rapidly close on any of a number of turbine trip initiation signals. The sequence of events for the turbine trip AOO is similar to the LOEL (Section 15.2.1) except that the steam flow following a turbine trip transient is isolated by closure of the main turbine stop valves rather than the main turbine control valves. The application states that because the LOEL event was analyzed by assuming an instantaneous cessation of both steam flow and feedwater flow from their full value (100 percent) at the beginning of the transient, the LOEL analysis bounds the turbine trip event and therefore the results and conclusions of Section 15.2.1 are also applicable for the turbine trip transient.

DCD Section 15.2.3 Loss of Condenser Vacuum

The application states that loss of condenser vacuum is one of the initiators that lead to a turbine trip which, as discussed in Section 15.2.2, is bounded by the analysis in Section 15.2.1.

The loss of condenser vacuum transient is therefore also bounded by the analysis of Section 15.2.1.

DCD Section 15.2.4 Closure of Main Steam Isolation Valve

The application states that inadvertent closure of the MSIVs would lead to a turbine trip which, as discussed in DCD Section 15.2.2, is bounded by the analysis in DCD Section 15.2.1. The closure of the MSIVs is therefore also bounded by the analysis of DCD Section 15.2.1.

DCD Section 15.2.5 Steam Pressure Regulator Failure

This event is not applicable to the US-APWR because it has no steam pressure regulators whose malfunction or failure could result in a steam flow transient.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.2.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Sections 15.2.1-15.2.5 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Sections 15.2.1-15.2.5 of NUREG-0800.

1. GDC 10, as it relates to the RCS design with appropriate margin so that SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to the design of the RCS and its auxiliaries with appropriate margin so that the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 17, as it relates to onsite and offsite electric power systems so that safety-related SSCs function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so that SAFDLs and RCPB design conditions are not exceeded during AOOs.

5. GDC 26, as it relates to the control of reactivity changes so that SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum DNBR remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations and by satisfaction of any other SAFDL applicable to the particular reactor design.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements in RG 1.105 are used for their impact on the plant response to the type of AOOs addressed in this section.

5. The most limiting plant system single failure, as defined in 10 CFR Part 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

6. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY-77-439, "Single Failure Criterion"; SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non Safety Systems"; and RG 1.206.

15.2.1.4 Technical Evaluation

Loss of External Load
The staff confirmed the LOEL analysis was performed using the MARVEL-M computer code and methods described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

The LOEL event was initiated by assuming an instantaneous step decrease (from 100 percent to 0 percent) in steam flow and feedwater flow. An instantaneous drop in steam flow at time zero is bounding for secondary side pressure because it ignores that steam is typically released during valve closure. Cessation of feedwater flow is conservative because continued feedwater flow would condense steam within the SGs, thereby lowering the calculated secondary side peak pressure.

Three cases of the LOEL event are analyzed (upon incorporation of **Confirmatory Item 15.00-2**). The first case was designed to calculate the DNBR, the second was designed to maximize primary pressure, and the third was designed to maximize secondary pressure. The initial conditions on reactor power, RCS temperature, and RCS pressure were found to be appropriate for each case as discussed in Section 15.0.0.4 of this SER.

All cases assumed 10 percent of the SG tubes were plugged. The staff felt this was non-conservative for the secondary side overpressure analysis, and issued RAI 789-5920, Question 15.02.01-15.02.05-10 asking for justification. In a response dated September 30, 2011, the applicant concluded SG tube plugging is not a key parameter for decrease in heat removal transients based on the results of a sensitivity analysis that assumed 0 percent SG tube plugging for the LOEL event. The staff finds the response acceptable because it demonstrated

there was no discernable difference in maximum primary or secondary side pressure between the sensitivity and DCD cases.

All cases assumed the minimum moderator density feedback. This is an appropriate choice to minimize the negative reactivity insertion as the temperatures of the moderator increase during this heatup event. Minimum values for Doppler feedback were used, as justified in the applicant's response to RAI 786-5881, Question 15.0.0-30, discussed in Section 15.0.0.4 of this SER.

The first case assumes the nonsafety-related pressurizer spray system is available to reduce RCS pressure and minimize DNBR. The pressurizer spray system is not modeled in the remaining cases, which are designed to maximize primary and secondary pressure. All cases assume manual rod control because in automatic rod control the RCCAs would be inserted to decrease power before the reactor trip occurs. The staff agrees this approach is consistent with the applicant's statement that non-safety systems are only assumed operational if they adversely impact the results.

The RTS, pressurizer safety valves and MSSVs are credited to mitigate this transient. For all cases, the reactor trips on the high pressurizer pressure signal and the time delays associated with this action are consistent with DCD Table 15.0-4. The setpoints associated with the pressurizer safety valves setpoints are consistent with the TS for all cases, but only the third case accurately models the MSSV setpoints. This is acceptable because the only acceptance criterion affected by MSSV setpoints is the secondary pressure, which the third case is designed to maximize. A single failure was assumed in one train of the RTS, but this has no impact on the safety analysis because any one of the remaining trains is adequate to provide the trip function. This is appropriate because it is not necessary to assume single failures in the passive spring-loaded pressurizer safety valves and MSSVs.

The LOEL event credits a reactor trip; therefore, in addition to the limiting single failure, the analysis assumes LOOP occurs 3 seconds after the reactor/turbine trip. The applicant states it is not necessary to run a separate LOOP case because the no-LOOP case is bounding with respect to the relevant acceptance criteria (DNBR, primary system pressure and secondary system pressure). The staff agrees, as discussed in Section 15.0.0.4 of this SER.

Numerical results for the minimum DNBR, primary system pressure and secondary system pressure are included in the applicant's response to RAI 297-2287, Question 15.0.0-16. The analysis results show that, while LOEL is the most limiting Chapter 15 event with respect to secondary system pressure, no acceptance limits are exceeded. No fuel failures are predicted and thus the radiological consequences for this event are bounded by the radiological consequences for the feedwater line break, which is reviewed under Section 15.2.8 of this SER.

<u>Turbine Trip</u>
The steam flow in the turbine trip event is isolated by closure of the main turbine stop valves whereas the steam flow in the LOEL event is isolated by closure of the main turbine control valves. Because the turbine stop valves close faster than the turbine control valves, the turbine trip generally results in a more severe transient. However, because the LOEL event takes no credit for valve closure time, the applicant stated the LOEL evaluation conservatively bounded the turbine trip event.

The staff agrees this is true for a turbine trip event with offsite power available, but notes that the LOEL evaluation, which assumes LOOP occurs 3 seconds after the reactor trip on high

pressurizer pressure, does not capture the sequence of events where a LOOP occurs 3 seconds after an initiating turbine trip. The staff issued RAI 303-2329, Question 15.2-2, asking for this evaluation. In a response dated July 3, 2009, the applicant stated, for an initiating turbine trip, the event could be mitigated by crediting the reactor trip on turbine trip (which is not a safety function, but is designed to be highly reliable in accordance with DCD Section 7.2.1.4.8). If the reactor trip on turbine trip is ignored and LOOP occurs 3 seconds after the turbine trip, the applicant stated that an evaluation showed the resulting DNBR for this event is bounded by the complete loss of forced reactor flow event in Section 15.3.1.2. The response also stated that the primary and secondary system pressures for a LOOP simultaneous with an initiating turbine trip (which ignores the 3 second delay) remain bounded by the DCD case. Due to the limited information presented regarding the supporting analyses, the staff was unable to conclude that the SRP criteria regarding LOOP were met and RAI 789-5920, Question 15.02.01-15.02.05-9, was issued requesting that an evaluation of a turbine trip with LOOP be added to the DCD. In a response dated September 30, 2011, the applicant proposed revising the DCD to include a turbine trip with LOOP analysis, and this case is evaluated below.

The turbine trip with LOOP analysis was performed using the MARVEL-M and VIPRE-01M computer codes and methods described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

Three cases of the turbine trip event were analyzed. The first case was designed to calculate the DNBR, the second was designed to maximize primary pressure, and the third was designed to maximize secondary pressure. The cases use the same initial conditions (power, RCS temperature, RCS pressure, SG tube plugging, feedback and non-safety system availability) as the LOEL cases, and the staff finds this appropriate because of the similarities between these heatup events.

The RTS, pressurizer safety valves and MSSVs are credited to mitigate this transient. The reactor will trip on either the high pressurizer pressure signal or the low RCP speed signal. The time delays used for these signals are consistent with DCD Table 15.0-4. The setpoints and single-failure analysis are identical to the LOEL analysis, which the staff finds appropriate because of the similarities between these heatup events.

For each case, the applicant performed sensitivity studies to determine the most adverse timing for LOOP, and found it was the time that caused the high pressurizer pressure and low RCP speed reactor trips to occur at the same time. The staff agrees this is a limiting approach because it results in the longest delay to reactor trip, which produces the largest heatup.

Numerical results for the minimum DNBR, primary system pressure and secondary system pressure are included in the applicant’s response to RAI 297-2287, Question 15.0.0-16. The analysis results show that, while the turbine trip is the most limiting Chapter 15 event with respect to primary system pressure, no acceptance limits are exceeded. No fuel failures are predicted; therefore, the radiological consequences for this event are bounded by the radiological consequences for the feedwater line break, which is reviewed under Section 15.2.8 of this SER.

As explained above, the staff finds the DCD changes proposed in response to RAI 789-5920, Question 15.02.01-15.02.05-9, acceptable and will use **Confirmatory Item 15.02-1** to track their incorporation into the next DCD revision.

Loss of Condenser Vacuum
A loss of condenser vacuum can result in a turbine trip and in a feedwater pump trip (on low suction pressure). Because the LOEL event assumes instantaneous loss of steam and feedwater at time zero, the staff agrees it bounds this event.

Closure of MSIV
Inadvertent closure of the MSIVs would lead to a turbine trip. Because the MSIVs are upstream of the turbine stop valves, this event may lead to a more severe secondary system pressure if the volume of the steam lines are included in the analysis. However, the MARVEL-M model used in the LOEL evaluation conservatively neglects the steam line volume; therefore, the staff agrees the LOEL event bounds inadvertent closure of the MSIV.

Steam Pressure Regulator Failure
The steam pressure regulator failure is not applicable to the US-APWR because steam pressure regulators are only used in BWRs.

15.2.1.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.2.1.6 Conclusions

With the exception of the confirmatory item discussed in this section, the staff concludes that the analyses of transients discussed in this section are acceptable and meet the requirements of GDCs 10, 13, 15, 17 and 26. This conclusion is based upon the following:

- The staff has determined that the applicant's analysis was performed using a mathematical model that was found acceptable as discussed in Section 15.0.2 of this SER. The parameters used as input to this model were reviewed and found to be suitably conservative and consistent with the plant design. In addition, the staff has determined that the positions of RG 1.53 as related to the single failure criterion and Regulatory Guide 1.105 for instruments have also been satisfied.

- The applicant has met the requirements of GDCs 10, 17, and 26 with respect to demonstrating that the resultant fuel integrity is maintained since the SAFDLs were not exceeded for this event, including that the minimum DNBR is greater than the 95/95 limit.

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of automatic protection systems occurred at values of monitored parameters that were within the instruments’ prescribed operating ranges. No manual protection systems are credited.

- The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by these events and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

- The applicant has met the requirements of GDCs 17 and 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the SAFDLs were not exceeded.

15.2.2 Turbine Trip

Review of this section of the DCD is documented under Section 15.2.1 of this SER.

15.2.3 Loss of Condenser Vacuum

Review of this section of the DCD is documented under Section 15.2.1 of this SER.

15.2.4 Closure of Main Steam Isolation Valve

Review of this section of the DCD is documented under Section 15.2.1 of this SER.

15.2.5 Steam Pressure Regulator Failure

This section of the DCD does not apply to the US-APWR because it has no steam pressure regulators.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

15.2.6.1 Introduction

The loss of non-emergency ac power (LNEP) is assumed to result in the loss of all power to the station auxiliaries. The causes are a complete loss of the external (offsite) grid accompanied by a turbine-generator trip or loss of the onsite ac distribution system. This event differs from the LOEL event considered in Section 15.2.1 because, in LOEL, ac power remains available to operate the station auxiliaries. In the LNEP transient, all the reactor coolant pump motors are de-energized simultaneously by the initiating event, resulting in a flow coast-down as well as a decrease in heat removal by the secondary system. This event is classified as an AOO.

15.2.6.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.2.6, summarized here as follows:

The loss of ac power has the following effects: simultaneous tripping of all RCPs, fast closure of turbine control valves, loss of feedwater due to loss of power to the condensate and feedwater

pumps and loss of condenser vacuum. After the RCPs trip, the core flow is reduced, increasing the RCS temperature and pressure. For convenience of the analyst, the event is initiated with loss of main feedwater and the loss of ac power is assumed to be coincident with a reactor trip on low SG water level. The GTGs are automatically started to provide electric power to vital loads. The sensible and decay heat loads are handled by actuation of the MSSVs and EFWS.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.2.6.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.2.6 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.2.6 of NUREG-0800.

1. GDC 10, as it relates to the RCS design with appropriate margin so that SAFDLs are not exceeded during normal operation including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so that the pressure boundary is not breached during normal operation including AOOs.

4. GDC 26, as it relates to reliable control of reactivity changes so that SAFDLs are not exceeded in AOOs. This control is accomplished by appropriate margin for malfunctions like stuck rods.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum DNBR remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations and by satisfaction of any other SAFDL applicable to the particular reactor design.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. For the requirements of GDC 10 and 15, the positions of RG 1.105, “Instrument Setpoints for Safety Related Systems,” have impact on the plant response to the type of AOOs addressed in this section.

5. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

15.2.6.4 Technical Evaluation

The staff confirmed the LNEP analysis was performed using the MARVEL-M computer code and methods described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

This transient is initiated with the loss of main feedwater flow while the reactor is at hot full power. When the SG water level reaches the low setpoint, a reactor trip is initiated. The LNEP (with RCP coastdown) is assumed to occur at the same time as the reactor trip. The staff agrees this is conservative because it maximizes the pressurizer water volume at the start of the LNEP event.

Two cases of the LNEP event are analyzed (upon incorporation of **Confirmatory Item 15.00-1**) The cases are identical except that in the first case the pressurizer water level is initiated at the nominal value plus uncertainty, while in the second case it is set to the maximum level allowed by TS 3.4.9. The second case is intentionally designed to show that the LNEP event will not lead to a more severe accident by demonstrating the pressurizer will not overfill and relieve liquid or two-phase flow through the pressurizer safety valves (which are only qualified for steam discharge). The initial conditions (reactor power, RCS temperature and RCS pressure) are based on nominal values with uncertainties added in the direction to maximize pressurizer water volume and were justified in the applicant's response to RAI 297-2287, Question 15.0.0-8.

The analysis uses the minimum moderator density feedback, which is appropriate because it minimizes the negative reactivity insertion as the density of the moderator decreases. The maximum value for Doppler feedback for this event was justified in the applicant's response to RAI 786-5881, Question 15.0.0-30, discussed in Section 15.0.0.4 of this SER.

The RTS, pressurizer safety valves, MSSVs, EFWS and GTGs are credited to mitigate this transient. For both cases, the low SG water level signal trips the reactor and initiates EFW. The time delays associated with these actions are consistent with DCD Tables 15.0-4 and 15.0-5. While the pressurizer safety valve setpoints are consistent with the TS, the MSSV setpoints are not. This is acceptable because the LNEP event is not limiting with respect to secondary pressure. The single failure assumed in this transient was loss of an EFWS train (from either failure of an EFWS component or failure of a GTG). The staff agrees with this assessment because a single failure of the RTS will not result in loss of function and it is not necessary to assume single failures in the passive spring loaded pressurizer safety valves and MSSVs.

DCD Figure 15.2.6-4, "Pressurizer Water Volume versus Time, Loss of Non-Emergency AC Power to the Station Auxiliaries," demonstrates that the pressurizer water volume remains well below the pressurizer capacity. As shown in Figure 16.298-1 "Pressurizer Water Volume vs. Time, Loss of Non-Emergency AC Power," included in the response to RAI 399-2992, Question 16-298, there is sufficient margin between the predicted water level and the pressurizer safety valves, which are located very near the top of the pressurizer.

The applicant stated that DNBR is not presented for this event because it is bounded by the complete loss of flow event (DCD Section 15.3.1.2). The staff requested a plot of DNBR in

order to confirm this assertion in RAI 304-2330, Question 15.2.6-1. The applicant responded on June 16, 2009, with a plot of DNBR versus time and the staff agrees that the DNBR limit is not challenged by the LNEP event. The minimum DNBR from this plot was included in the summary of results provided in the applicant's response to RAI 297-2287, Question 15.0.0-16 (discussed in Section 15.0.0.4 of this SER). The LNEP results show that no acceptance limits are exceeded. No fuel failures are predicted; therefore, the radiological consequences for this event are bounded by the radiological consequences for the feedwater break, which is reviewed under Section 15.2.8 of this SER.

15.2.6.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.2.6.6 Conclusions

The staff concludes that, pending closure of Confirmatory Item 15.00-1, that the plant design as to transients expected to occur with moderate frequency and to result in the loss of all power to the station auxiliaries is acceptable and meets the relevant requirements of GDCs 10, 13, 15, and 26 and the applicable TMI Action Plan items. This conclusion is based on the following findings:

- The applicant meets the requirements of GDCs 10 and 26 by demonstrating that fuel integrity is maintained because the SAFDLs were not exceeded for the event.

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

- The applicant meets GDC 15 requirements by demonstrating that the reactor coolant pressure boundary limits were not exceeded by this event and that resultant leakage is within acceptable limits. These requirements are met because the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressure.

- The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during this event with appropriate margin for stuck rods because the SAFDLs were not exceeded.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Introduction

A loss-of-normal-feedwater-flow event (LOFW) could occur from pump failures, valve malfunctions, or LOOP. The LOFW results in a reduction of the secondary system's ability to remove heat generated by the reactor core. As a result, the reactor coolant temperature and

pressure will increase, which eventually requires a reactor trip to prevent fuel damage. This event is classified as an AOO.

15.2.7.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.2.7, summarized here as follows:

This section addresses the LOFW caused by pump failures and valve malfunctions. When normal feedwater is lost, the water level in the SGs drop as the remaining water inventory is boiled off. A low SG water level signal will trip the reactor and initiate the EFWS. After the trip, steam produced from decay heat and sensible heat is relieved through the MSSVs to maintain the plant at hot-standby conditions. A LOOP initiated loss of normal feedwater flow is addressed in the LNEP analysis of DCD Section 15.2.6.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.2.7.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.2.7 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.2.7 of NUREG-0800.

1. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. 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.

4. GDC 17, as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function during normal operation, including AOOs. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that acceptable fuel design limits and design conditions of the RCPB are not exceeded during an AOO.

5. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

6. 10 CFR 50.34(f)(1)(ii), and 10 CFR 50.34(f)(2)(xii) as they relate to the performance requirements of the EFWS for the LOFW event.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum DNBR remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations and by satisfaction of any other SAFDL applicable to the particular reactor design.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. For the requirements of GDC 10 and 15, the positions of RG 1.105, "Instrument Setpoints for Safety Related Systems," have impact on the plant response to the type of AOOs addressed in this section.

5. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

6. The guidance provided in SECY 77-439, "Single Failure Criterion," SECY 94-084, "Policy and Technical Issues associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," and RG 1.206 with respect to the consideration of the performance of non-safety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

7. TMI Action Plan Item II.K.2.19 of NUREG 0737, "Clarification of TMI Action Plan Requirements," as it relates to the performance requirements of the EFWS for the LOFW event.

15.2.7.4 Technical Evaluation

The staff confirmed the LOFW analysis was performed using the MARVEL-M computer code and methods documented in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

Four cases of the LOFW event are analyzed (upon incorporation of Confirmatory Item 15.00-1). The first case was designed to calculate the DNBR, the second was designed to maximize primary pressure, and the remaining two were designed to maximize pressurizer water level. The last two cases are identical except that in the third case, the pressurizer water level is initiated at the nominal value plus uncertainty while in the last case, it is set to the maximum level allowed by TS 3.4.9. The last case is intentionally designed to show that the LOFW event will not lead to a more severe accident by demonstrating the pressurizer will not overfill and relieve liquid or two-phase flow through the pressurizer safety valves (which are only qualified for steam discharge).

Because the DNBR calculations use the RTDP, nominal values are used for reactor power, reactor coolant average temperature, and RCS pressure for the first case. The staff finds this appropriate, as discussed in Section 15.0.0.4 of this SER.

For the second case, uncertainties are added to the nominal conditions in the direction to maximize RCS pressure and these values were justified in the applicant's response to RAI 297-2287, Question 15.0.0-8. The final cases added uncertainties to the nominal conditions in the direction to maximize pressurizer water level. The applicant's response to RAI 297-2287, Question 15.0.0-8, indicates that, if the uncertainties on initial power and RCS temperature were added in the direction opposite to what was used in the DCD, the pressurizer water level would be slightly higher. This is acceptable because, even with this increase, the pressurizer water level in the LOFW event remains bound by the LNEP event (DCD Section 15.2.6).

All LOFW transients are modeled with the minimum moderator density feedback. This is an appropriate choice because it minimizes the negative reactivity insertion as the temperatures of the moderator increase. The maximum values were used for Doppler feedback. The applicant's response to RAI 786-5881, Question 15.0.0-30, demonstrates that, if the minimum values were used, the DNBR, RCP outlet pressure and pressurizer water level would all be slightly more limiting. The staff finds that the parameters used in the DCD analysis are acceptable because even with these slight increases, the LOFW event remains bound by turbine trip (DCD Section 15.2.2) for DNBR and RCP outlet pressure. Additionally, the pressurizer water level in the LOFW event remains bound by the LNEP event (DCD Section 15.2.6) even if the increase due to Doppler feedback is combined with the increase due to initial conditions.

The nonsafety-related pressurizer spray system is available for all cases except the second, which is designed to maximize RCS pressure. The nonsafety-related pressurizer heater is available for all cases but the first, which is designed to minimize DNBR. The staff agrees this approach is consistent with the DCD Section 15.0.0.4 discussion where non-safety systems are assumed operational only if they adversely impact the results.

The RTS, pressurizer safety valves, MSSVs, and EFWS are credited to mitigate this transient. In each case, the low SG water level signal trips the reactor and initiates EFW. The time delays associated with these actions are consistent with DCD Tables 15.0-4 and 15.0-5. While the pressurizer safety valve setpoints are consistent with the TS, the MSSV setpoints are not. This is acceptable because the LOFW event is not limiting with respect to secondary pressure. The single failure assumed in this transient was loss of an EFWS train. The staff agrees with this assessment because a single failure of the RTS will not result in loss of function and it is not necessary to assume single failures in the passive spring-loaded pressurizer safety valves and MSSVs.

The applicant stated that it is not necessary to present a transient of LOFW in combination with LOOP because that event is addressed by the LNEP event presented in DCD Section 15.2.6. The staff agrees because if LOOP is considered with the LOFW event, the RCPs would begin to coast down 3 seconds after the reactor trip. The LNEP event, which was initiated with loss of normal feedwater, assumes the RCPs begin coastdown concurrent with the reactor trip; therefore, it bounds LOFW with LOOP.

DCD Figure 15.2.7-11, "Pressurizer Water Volume versus Time, Loss of Normal Feedwater Flow," demonstrates that the pressurizer water volume remains well below the pressurizer

capacity, and is less limiting than the LNEP event. Numerical results for the minimum DNBR, primary system pressure and secondary system pressure are included in the applicant’s response to RAI 297-2287, Question 15.0.0-16, demonstrating that no acceptance limits are exceeded.

No fuel failures are predicted; therefore, the radiological consequences for this event are bounded by the radiological consequences of the feedwater line break, which is reviewed under Section 15.2.8 of this SER.

15.2.7.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.2.7.6 Conclusions

The staff concludes that the plant design is acceptable with regard to transients resulting from LOFW and that the predicted response meets the requirements of GDCs 10, 13, 15, 17, and 26. This conclusion is based on the following:

- The staff has determined that the applicant's analysis was performed using a mathematical model that was found acceptable as discussed in Section 15.0.2 of this SER. The parameters used as input to this model were reviewed and found to be suitably conservative and consistent with the plant design. In addition, the staff has determined that the positions of RG 1.53 as related to the single failure criterion and RG 1.105 for instruments have also been satisfied.

- The applicant has met the requirements of GDCs 10, 17, and 26 with respect to demonstrating that the resultant fuel integrity is maintained, since the SAFDLs were not exceeded for this event, including that the minimum DNBR is greater than the 95/95 limit.

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of automatic protection systems occurred at values of monitored parameters that were within the instruments’ prescribed operating ranges. No manual protection systems are credited.

- The applicant has met the requirements of GDC 15 with respect to demonstrating that the RCPB limits have not been exceeded by these events and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

- The applicant has met the requirements of GDCs 17 and 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the SAFDLs were not exceeded.

15.2.8 Feedwater System Pipe Break Inside and Outside Containment

15.2.8.1 Introduction

A feedwater system pipe break causes a loss of inventory from the saturated liquid mass in the SG resulting in RCS heatup and pressurization. Minor feedwater system pipe breaks are classified as AOOs. Major feedwater pipe breaks, which are defined as those large enough to prevent the addition of sufficient feedwater to maintain the SG inventory are classified as PAs.

15.2.8.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 safety analysis in Section 15.2.8, summarized here as follows:

A feedwater system pipe break reduces the ability to remove heat generated by the core because feedwater flow to the SGs is reduced, fluid in the SGs may be discharged through the break and no longer available for heat removal, and the break may be large enough to prevent the addition of feedwater after the trip.

Minor feedwater breaks that result in continued feedwater addition at a rate insufficient to maintain SG level are bound by the LOFW transient evaluated in DCD Section 15.2.7. The most limiting feedwater break is the double-ended rupture of the feedwater line between the main feedwater check valve and SG. A break at this location results in a rapid blowdown of one SG through the ruptured piping. The EFWS train that would normally supply the broken loop will also spill out through the ruptured piping. For convenience of the analyst, the event is initiated with loss of main feedwater and the feedwater break is assumed to be concurrent with reaching a reactor trip setpoint on low SG water level. LOOP is also assumed to be concurrent with the reactor trip, and GTGs are automatically started to provide electric power to the EFWS. The event is mitigated by a reactor trip, actuation of pressurizer safety valves and MSSVs, isolation of EFW to the failed SG, and actuation of EFW to the intact SGs.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.2.8.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.2.8 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.2.8 of NUREG-0800.

1. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. GDC 17, as it relates to onsite and offsite electric power systems for safety-related SSCs to function. The safety function for each power system (assuming the other system is not functioning) must be of sufficient capacity and capability so that design conditions of the RCPB are not exceeded and the core is cooled in PAs.

3. GDC 27 and GDC 28, as it relates to the RCS design with appropriate margin so that SAFDLs are not exceeded and core cooling capability is maintained.

4. GDC 31, as it relates to the RCS design with sufficient margin so that the boundary is non-brittle and the probability of fracture propagation is minimized.

5. GDC 35, as it relates to the design of the RCS and its auxiliaries for abundant emergency core cooling.

6. 10 CFR 100, as it relates to calculated doses at the site boundary.

Acceptance criteria adequate to meet the above requirements include:

1. Requirements for maintenance of adequate decay heat removal by the EFWS are in 10 CFR 50.34(f)(1)(ii), (TMI issue II E 1.1) and 10 CFR 50.34(f)(2)(xii), (TMI issue II E1.2). Requirements for RCP operation are in 10 CFR 50.34(f)(1)(iii), (TMI issue 2 K 2).

2. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.

3. The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations. If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an acceptable fuel damage model including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.

4. Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 guidelines.

5. The integrity of the RCPs should be maintained so that the loss of alternating current power and containment isolation do not result in seal damage.

6. The AFWS must be safety grade and automatically initiated when required.

15.2.8.4 Technical Evaluation

The staff confirmed the feedwater break evaluation was performed using the MARVEL-M computer code and methods documented in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

This transient is initiated with the loss of main feedwater flow while the reactor is at HFP. When the SG water level reaches the low setpoint, a reactor trip is initiated, concurrent with a feedwater break. The staff agrees this is a conservative approach because the break occurs at the time of minimum SG inventory, which maximizes the RCS heat up. Three sets of initial conditions are evaluated, designed to maximize primary pressure, hot leg boiling, and pressurizer water volume.

In each case, uncertainties are added to the nominal conditions for power, RCS temperature and RCS pressure in the direction to maximize the key parameter. The direction of each uncertainty was justified by sensitivity studies provided in the applicant's response to RAI 297-2287, Question 15.0.0-8. All cases assumed the minimum moderator density feedback, which is appropriate because it minimizes the negative reactivity insertion as the temperatures of the moderator increase. The maximum values were used for Doppler feedback, as justified by the applicant's response to RAI 786-5881, Question 15.0.0-30. The staff reviewed assumptions regarding the control systems modeled in each case and agrees the control system was only included if it made the parameter more severe.

The pressurizer water level is initiated at the nominal level plus uncertainty rather than the maximum level allowed by TS 3.4.9. As described in Section 15.0.0.4 of this SER, the staff agrees this is acceptable because it supports the design requirement from Section 5.4.10.1 (upon incorporation of **Confirmatory Item 15.00-1**) that there is no water relief through the pressurizer safety valves for this event when initiated from an initial pressurizer water level that is less than or equal to the nominal plus instrument uncertainty.

For all three cases, the reactor trip on low SG water level initiates a turbine trip and concurrent LOOP/RCP coastdown. No credit is given for the assumed 3-second delay between turbine trip and LOOP (DCD Section 15.0.0.4) and the feedwater line break sensitivity study included in the response to RAI 297-2287, Question 15.0.0-3, demonstrates this is conservative. The staff agrees that only the LOOP case needs to be presented because the reduction in RCS flow and additional time needed to start the EFWS pumps increases the severity of the heatup, making it more limiting than no-LOOP.

In all cases, the accident is mitigated by reactor trip, opening of safety valves in the pressurizer and SG, actuation of EFW to the intact SGs and isolation of EFW to the faulted SG. The low SG water level signal trips the reactor and initiates EFW and the low main steam line pressure signal isolates EFW. The time delays associated with these actions are consistent with DCD Tables 15.0-4 and 15.0-5. While the pressurizer safety valve setpoints are consistent with the TS, the MSSV setpoints are not. This is acceptable because the feedwater break event is not limiting with respect to secondary pressure. The single failure is assumed to be one EFWS train. Because the EFWS train supplying the faulted SG is assumed to spill from the break, only two of the four EFWS trains supply feedwater to the system. This is limiting because single failure of the RTS or EFW isolation valves will not result in loss of function and it is not necessary to assume single failure in the MSSVs and pressurizer safety valves.

The feedwater line break case designed to maximize RCS pressure was one of the sample transient events included in MUAP-07010-P. As such, the staff evaluation of MUAP-07010-P includes additional details on the feedwater break methodology

Because DNBR plots were not included in the DCD, the staff requested them in RAI 305-2331 Question 15.2.8-3. In a response dated July 3, 2009, the applicant presented the DNBR transient for a case that differed from the three included in the DCD. The case in the RAI

response assumed nominal initial conditions, which the staff accepts because DNBR is analyzed with the RTDP, an NRC-approved methodology that uses nominal operating conditions and accounts for relevant uncertainties via a statistical procedure.

The DNBR case described in the RAI response also assumed a 3-second delay between the turbine trip and LOOP (as stated in the July 3, 2009 response to RAI 305-2331, Question 15.2.8-1), which was found to be acceptable in Section 15.0.0.4 of this SER. The resulting plot demonstrated considerable margin to the analytical limit. The minimum DNBR from this plot was included in the summary of results provided in the applicant's response to RAI 297-2287, Question 15.0.0-10 (discussed in Section 15.0.0.4 of this SER). Even though this event is classified as a PA, the more limiting AOO acceptance criteria are met for minimum DNBR, primary system pressure and secondary system pressure. The staff notes that this is the limiting Chapter 15 event with respect to primary pressure.

While no fuel failures are predicted, radiation could be released to the environment in the case where there is primary-to-secondary leakage from normal plant operations. As explained in a July 3, 2009 response to RAI 303-2329, Question 15.2-8, the feedwater line break is similar to the main steam system break (Section 15.1.5) because both events result in the complete blowdown of the affected loop to the atmosphere. Both events also assume decay and sensible heat removal from the intact loops is released to the atmosphere through the secondary safety and relief valves. Neither event predicts fuel failure; therefore, primary source term and primary-to-secondary leakage assumptions are the same. However, because the SG inventory is larger at hot zero power (when main steam system break is initiated) than at hot full power (when feedwater line break is initiated), the main steam system break will bound the feedwater line break with respect to radiological consequences. The staff agrees that the MSLB bounds the feedwater break because it has a larger steam release. The radiological consequences of a MSLB are evaluated in Section 15.0.3.4 of this SER.

15.2.8.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.2.8.6 Conclusions

The staff concludes that the applicant's analysis of consequences of postulated feedwater line breaks meets the requirements of GDCs 13, 17, 27, 28, 31, and 35 for ability to insert control rods and ability to cool the core, 10 CFR 100 guidelines for radiological doses at the site boundary, and applicable Three Mile Island Action Plan Items. This conclusion is based on the following findings:

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of automatic protection systems occurred at values of monitored parameters that were within the instruments' prescribed operating ranges. No credit is taken for actuation of manual protection systems.

- The applicant meets GDCs 27 and 28 requirements by demonstrating the minimum DNBR remains above the 95/95 limit. Hence, no fuel failures are predicted, demonstrating maintained ability to insert the control rod and no loss of core cooling capability.

- The applicant meets GDC 31 requirements for demonstrating primary system boundary capability to withstand the PA. The maximum RCS pressure was below 110 percent of the design pressure.

- The applicant meets GDC 35 requirements for demonstrating emergency cooling system adequacy for abundant core cooling and reactivity control (via boron injection).

- The analyses of effects of feedwater line break accidents inside and outside containment during various modes of operation with and without offsite power have been reviewed and evaluated by a mathematical model as discussed in Section 15.0.2.4 of this SER.

- The input parameters for this model were reviewed and found suitably conservative.

- The radioactivity release is bounded by the evaluation of the MSLB event discussed in Section 15.1.5 of this SER.

- The applicant meets 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) requirements for demonstrating the adequacy of the EFWS design to remove decay heat following feedwater piping failures.

- Section 15.6.5 of this SER describes how the applicant meets 10 CFR 50.34(f)(1)(iii) requirements for demonstrating RCP seal capability to withstand the PA.

15.3 Decrease in Reactor Coolant System Flow Rate

15.3.1 Loss of Forced Reactor Coolant Flow – Trip of Pump Motor

15.3.1.1 Introduction

A decrease in reactor coolant flow while a plant is at power could result in degraded core heat transfer. An increase in fuel temperature and accompanying fuel damage then could result if SAFDLs are exceeded during the transient. This section covers a number of transients expected to occur with moderate frequency that decrease forced reactor coolant flow.

15.3.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.3.1, summarized here as follows:

The application describes analyses that have been performed for events that could result in a decrease in RCS flow rate, which can lead to an increase in the primary coolant temperature.

Analyses of the following events are discussed in this section:

- Partial loss of forced reactor coolant flow, as described in DCD Section 15.3.1.1, and
- Complete loss of forced reactor coolant flow, which is described in DCD Section 15.3.1.2.

Loss of forced reactor coolant flow events can result from mechanical or electrical failures in one or more RCPs or from a fault in the power supply to the pump motor. A partial loss of coolant flow accident results from a simultaneous loss of electrical power to one or more of the four RCPs. The complete loss of forced reactor coolant flow is initiated by malfunctions that cause the loss of electrical power or the decrease of offsite power frequency to all four RCPs during power operation, resulting in a reduction in the core cooling capabilities. If the reactor is at power at the time of these transients, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature and a decrease in minimum DNBR. The partial or complete loss of flow events are terminated by the low reactor coolant flow trip or by the low RCP speed trip, which prevents DNB occurrence. The limiting single failure for the events is the loss of one train of the RTS. Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Further details about the RTS are provided in DCD Section 7.2, "Reactor Trip System." Both partial and complete loss of flow events are classified as AOOs, as described in DCD Section 15.0.0.1.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.3.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.3.1-15.3.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.3.1-15.3.2 of NUREG-0800.

1. GDC 10 and GDC 20, as it relates to the design of the RCS with appropriate margin so that SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to design of the RCS and its auxiliaries to provide appropriate margin so that the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 17, as it relates to the onsite and offsite electric power systems so that SSCs important to safety function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) must be to provide sufficient capacity and capability so that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded during AOOs.

5. GDC 26, as it relates to the reliable control of reactivity changes so that SAFDLs are not exceeded, including during AOOs. This control is accomplished by accounting for appropriate margin for malfunctions (e.g., stuck rods).

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum DNBR remaining above the 95/95 DNBR limit based on acceptable correlations.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements in RG 1.105 are evaluated for their impact on the plant response to the type of AOOs addressed in this section.

5. Onsite and offsite electric power systems must be maintained so that safety-related SSCs function during normal operation and AOOs.

6. The most limiting plant system single failure, as defined in “Definitions and Explanations,” 10 CFR 50, Appendix A, must be assumed in the analysis, according to the guidance of RG 1.53.

7. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY-77-439, SECY-94-084, and RG 1.206.

15.3.1.4 Technical Evaluation

The NRC staff reviewed the loss of forced reactor coolant flow analyses described in the US-APWR DCD, in accordance with SRP Section 15.3.1-15.3.2. The acceptability of the system is based on meeting the requirements of GDC criteria and SRP acceptance guidance as described above in the regulatory basis. The partial and complete loss of forced reactor coolant flow event analyses are discussed in DCD Sections 15.3.1.1 and 15.3.1.2, respectively. In the subsections of these DCD sections, the applicant described the sequence of events and system operation, analyses of the events including evaluation model and results, radiological consequences, and conclusions of the analyses.

The particular events analyzed in this evaluation are partial and complete loss of reactor coolant flow. A decrease in reactor coolant flow while a plant is at power could result in degraded core heat transfer. An increase in fuel temperature and accompanying fuel damage then could result if SAFDLs are exceeded during the transient. A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor or a fault in the power supply to the pump motor. A complete loss of forced coolant flow may be the result of the simultaneous loss of electrical power or the decrease of offsite power frequency to all pump motors.

Partial loss of forced reactor coolant flow

The technical information of the partial loss of forced reactor coolant flow event is provided in the subsections of DCD Section 15.3.1.1, where the applicant described the sequence of events and system operation, analysis of the event, results of analysis, barrier performance, and radiological consequences. Further, DCD Table 15.3.1.1-1, "Time Sequence of Events for Partial Loss of Forced Reactor Coolant," lists the sequence and timing of major events, and Figures 15.3.1.1-1 "RCS Total and Loop Volumetric Flow versus Time, Partial Loss of Forced Reactor Coolant Flow" through 15.3.1.1-6 "DNBR versus Time, Partial Loss of Forced Reactor Coolant Flow," depict the plots of key system parameters versus time from the core and the system performance evaluation. A summary of technical information in the DCD is as follows:

The DCD states that a partial loss of forced reactor coolant flow can result from a mechanical or electrical failure in one or more RCPs or from a fault in the power supply to the pump motor. If the reactor is at power at the time of the transient, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature and a decrease in minimum DNBR. This transient is terminated by the low reactor coolant flow trip, which prevents DNB occurrence.

Complete loss of forced reactor coolant flow

The technical information of the complete loss of forced reactor coolant flow event is provided in the subsections of DCD Section 15.3.1.2, where the applicant described the sequence of events and system operation, analysis of the events, results of analysis, barrier performance, and radiological consequences for two separate loss of flow events. The first case models a loss of power supply with a flow coastdown curve based on the inertia of the RCP flywheel. The second case models a frequency decay where the RCPs coast down at the same linear rate that the frequency decreases.

DCD Table 15.3.1.2-1 "Time Sequence of Events for Loss of Power Supply Resulting in a Complete Loss of Forced Reactor Coolant Flow" lists the time sequence of events for loss of power supply, whereas DCD Table 15.3.1.2-2 "Time Sequence of Events for Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow" lists the time sequence of events for frequency decay. Further, Figures 15.3.1.2-1 "RCS Total Flow versus Time, Complete Loss of Forced Reactor Coolant Flow" through 15.3.1.2-12, "DNBR versus Time, Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow" show the transient responses for key parameters of the above events.

A complete loss of forced reactor coolant flow event results in a reduction in the core cooling capabilities. If the reactor is at power at the time of the transient, the immediate effect of a complete loss of coolant flow is a rapid increase in coolant temperature and decrease in minimum DNBR. As the pumps slow down, a reactor trip signal is generated by the low RCP speed trip. The rate of change in the flow is less severe for the loss of power supply case due to the inertia of the pump flywheels. In both cases, the flow decreases prior to the reactor trip, resulting in a decrease in the DNBR. The minimum DNBR occurs shortly after the reactor trip following the sharp decrease in power. Although the minimum DNBR is the lowest in the frequency decay case, it remains above the 95/95 DNBR design limit.

Staff Evaluation:

- Conformance to GDC 10 requires the design of the reactor coolant system with appropriate margins so that SAFDLs are not exceeded and the fuel-cladding integrity is maintained during normal operations and AOO including loss of forced-reactor coolant flow.

In order to meet the GDC 10 criteria for fuel-clad integrity, in RAI 306-2333, Question 15.3.1-7, the staff requested the applicant to provide the peak fuel centerline temperature as function of time for the reactor transients discussed in DCD Sections 15.3.1 through 15.3.4, and to explain the associated safety limit for the fuel centerline temperature. In a December 20, 2011, response, the applicant provided plots of the peak fuel centerline temperature as a function of time for the following events: partial loss of flow, loss of power supply resulting in a complete loss of flow, frequency decay resulting in a complete loss of flow, and the RCP rotor seizure (which is discussed in Section 15.3.3 of this SER). The applicant further stated that the AOO safety limit for fuel centerline temperature is set to be less than the fuel melting temperature. The difference between the safety limit and the melting temperature accounts for a 250°C [450°F] temperature uncertainty plus burnup effects calculated at the burnup that gives the minimum margin between the fuel temperature and the melting temperature. The staff finds that this approach is reasonable and that the associated plots exhibited considerable margin to the safety limit value. Therefore, the staff determined that the design of the RCS meets the GDC 10 criteria, as it relates to not exceeding the design margins and fuel centerline temperature and fuel-clad integrity during the loss-of-reactor-coolant-flow transients. Further, staff determined that the applicant's response to **RAI 306-2333, Question 15.3.1-7,** was acceptable and is therefore resolved.

- With respect to GDC 13, as it relates to the availability of instrumentation and controls to monitor and maintain the variables and systems to ensure safety during these reactor coolant transients, the staff evaluation is provided in Chapter 7 and Section 15.0.0.4 of this SER.

- Conformance to GDC 15, as it relates to maintaining the reactor coolant pressure boundary for AOO of loss of forced reactor coolant flow. Item 1, "SRP Acceptance Criteria," of SRP Section 15.3.1 – 15.3.2, Section II, provides that pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

In Sections 15.3.1.1.4 and 15.3.1.2.4, the DCD states that the loss of forced reactor coolant flow event does not result in exceeding any RCPB or containment volume fission product barrier design limits. The results of the core and system evaluation case demonstrate that RCS pressure remains well below 110 percent of system design pressure. Also, it states that the main steam pressure cannot challenge the main steam system pressure design limit, and therefore, the integrity of the reactor coolant and the main steam system pressure boundary are maintained.

However, since the information in the DCD is not adequate to evaluate this statement about the challenge to the main steam pressure limit, the staff requested additional information in RAI 306-2333, Question 15.3.1-4. In response, in a letter dated June 16, 2009, the applicant provided the SG pressure-vs.-time curves for the partial loss of flow, loss of power supply resulting in complete loss of flow, and frequency decay resulting in a complete loss of flow events as requested. The applicant further stated that the MSSV is modeled to lift at 103 percent of the SG design pressure. While this valve setting does not bound the three MSSV lift settings from TS 3.7.1, it is acceptable for use here because the Section 15.3.1 events are not

the limiting events for main steam line pressure. The two most limiting events (Sections 15.2.1 and 15.2.2) use a more accurate model of the MSSV as described in Section 15.0.0.4 of this SER. The staff reviewed the additional information and confirmed the statements in DCD Subsections 15.3.1.1.3.3 and 15.3.1.2.3.3 that the steam line pressures for these events are bounded by that of the RCP rotor seizure event which is less than 110 percent of the design pressure. Thus, the staff finds the applicant's response to RAI 306-2333, Question 15.3.1-4, acceptable, as it meets the SRP guidance for maintaining the main steam system pressure below 110 percent of the design pressure for the forced reactor coolant flow events.

- GDC 17, "Electrical Power Systems"

With respect to electric power sources, GDC 17 requires and guidance in Item 5, "SRP Acceptance Criteria," of SRP Section 15.3.1 – 15.3.2, Section II, recommends that onsite and offsite power systems be maintained with adequate capacity and capability, so that safety-related SSCs perform intended functions and the design conditions of the RCPB are not exceeded during normal operation and AOOs. In DCD Tier 2 Section 15.3.1.1.2 and 15.3.1.2.2, the applicant states that the RCPs are connected to separate plant buses and a failure of one plant bus does not cause two or more pumps to stop at the same time. During reactor operation, the buses are supplied with power from the generator. If the power from the generator is cut off, the buses are supplied with power from an offsite transmission line. The applicant further states that, although the US-APWR is configured for each pump to have its own electric power source, the partial loss of flow analysis conservatively assumes two pumps trip simultaneously. Thus, the analysis bounds the case where two pumps share a common source of power. Further, the evaluation of complete loss of reactor flow includes the effects of a postulated LOOP because this event can be initiated by a reactor trip. Based on the above discussion, and a review of the results of the analysis, the staff determined that the US-APWR design has adequate electric power sources to conform to the GDC 17 requirement and meets the associated SRP guidance.

- With respect to the most limiting single failure consideration, the SRP Acceptance Criteria in SRP Sections 15.3.1 – 15.3.2 states that this limiting condition must be assumed in the analysis and follow the guidance of RG 1.53. Further, as it relates to the single failure criteria, the guidance in SRP Section 15.3.1 - 15.3.2, Section III, Item 2, states that for new applications LOOP should not be considered a single failure; each loss of flow transient should be analyzed with and without a LOOP in conjunction with a single active failure.

In DCD Section 15.3.1.1.2, the applicant stated that the limiting single failure for these events is failure of one train of the RTS. The applicant further described that any one of the remaining trains is adequate to provide the protection functions credited in this assessment.

The staff has reviewed the applicant's limiting single failures and the methodology for the selection of the limiting single failures. The selection methodology is based upon an event-specific review of the progression of the event and the assumed mitigative equipment and its associated function. The limiting single failure for both "Loss of Forced Reactor Coolant Flow" transients was the failure of one train of the RTS. Also, determination of this limiting single failure for Chapter 15 transients, including those of loss of coolant events, is provided in the applicant's response to RAI 297-2287, Question 15.0.0-11.

However, the DCD does not provide adequate information to satisfy the above SRP criteria. Therefore, in RAI 306-2333, Question 15.3.1-1, the staff requested the results of the applicant's calculations that include the occurrence of a LOOP. In response to this RAI, in a letter dated

June 16, 2009, the applicant stated that the response to RAI 297-2287, Question 15.0.0-3, demonstrates the limiting DNBR occurs prior to RCP coastdown, thus, it is not necessary to perform a LOOP analysis. As further support, the applicant provided the results of a sensitivity study demonstrating the minimum DNBR from a partial loss of forced RCS flow with no LOOP bounds the case with LOOP assumed 3 seconds after the reactor/turbine trip. For the complete loss of forced reactor coolant flow, all of the RCPs trip as the initiating event of this transient; therefore, it was not required to consider a LOOP in this case. As described in Section 15.0.0.4 of this SER, the staff agrees that the no-LOOP case sufficiently captures the limiting DNBR and peak RCS and secondary side pressures. Therefore, the staff finds the design of the RCS and applicant's LOOP analysis for loss-of-reactor-coolant-flow transients meet the SRP guidance.

- GDC 26 "Reactivity Control System Redundancy and Capability"

The applicant's analysis conforms to GDC 26 because the analysis assumes the single highest-reactivity-worth RCCA remains fully withdrawn.

- The SRP acceptance criteria in SRP Section 15.3.1 – 15.3.2, recommend that parameter values in the analytical model should be suitably conservative.

In RAI 306-2333, Questions 15.3.1-2 and 15.3.1-3, the staff requested that the applicant provide additional information concerning the dependence of the minimum DNBR on the mixing factors, FMXI and FMXO, that were assumed in the calculations for the partial loss of forced reactor coolant flow event and the complete loss of forced reactor coolant flow event. In response to RAI 306-2333, Question 15.3.1-2, the applicant pointed out that in the case of partial loss of forced reactor coolant flow the RCS pressure and inlet temperature were kept constant in the DNBR calculation; therefore, the inlet and outlet mixing factors also have no effect on the minimum DNBR. The staff finds this is reasonable given the short duration of the event (minimum DNBR occurs in less than 4 seconds).

- SRP guidance in Sections 15.3.1 – 15.3.2, Subsection I, "Areas of Review," recommends that results of the applicant analyses are reviewed for whether values of pertinent system parameters are within expected ranges, including the minimum DNBR. Also, the guidance recommends that analytical methods are reviewed for whether the mathematical modeling and computer codes are reviewed and accepted by the staff.

The staff confirmed that the transients are simulated using the computer codes MARVEL-M and VIPRE-01M in accordance with the methodology described in MUAP-07010-P. NRC approval of MUAP-07010-P, including an evaluation of its restrictions and exceptions, and the applicable ranges of parameters when applied to this transient, is described in Section 15.0.2.4 of this SER.

The results of the analysis demonstrate that for a complete loss of forced reactor coolant flow transient, there will be a more rapid reduction in core cooling capability than for the partial loss of forced reactor coolant flow transient. However, the resulting transient does not cause the minimum DNBR to decrease below the 95/95 limit and no fuel failures are predicted. The RCS pressure and main steam pressure remain below 110 percent of their respective system design pressures; thus, the integrity of the reactor coolant and main steam system pressure boundaries are maintained. The staff concludes that the calculated results are in satisfaction of the specified acceptance criteria and are acceptable.

Additionally, the DCD states that the radiological consequences of these events are bounded by the radiological consequences of the RCP's rotor seizure, which is evaluated in Section 15.3.3.5 of this SER. The staff agrees because there were no fuel failures predicted for the loss-of-flow events. Therefore, based on the above discussion, the staff finds the applicant analysis for the loss-of-forced-reactor-coolant-flow transient acceptable.

15.3.1.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.3.1.6 Conclusions

Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. The complete loss of forced reactor coolant flow transient caused by a frequency decay was found to be the most limiting of these events for core thermal margins and pressure within the reactor coolant and main steam systems. The applicant analyzed this transient using a mathematical model described in Section 15.0.2.4 of this SER. The values of the input parameters to this model were reviewed and found suitably conservative.

The staff concludes that the plant design for transients expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow is acceptable and meets the requirements of GDCs 10, 13, 15, 17, 20, and 26. This conclusion is based on the following findings:

- The applicant meets the requirements of GDCs 10,20,13,15, 17, and 26 by demonstrating that SAFDLs are not exceeded in this event.

- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

- The applicant meets the requirements of GDCs 15 and 17 by demonstrating that the RCPB limits are not exceeded in this event. This requirement is met as the analysis shows that the maximum pressure of the reactor coolant and main steam systems does not exceed 110 percent of the design pressure.

- The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during this event with appropriate margin for stuck rods.

- The applicant meets the positions of RG 1.53, SECY 77-439, SECY 94-084, and RG 1.206 on the single-failure criterion and RG 1.105 on instrument actuations of safety-related SSCs.

15.3.2 Loss of Forced Reactor Coolant Flow – Flow Controller Malfunction

This section is not applicable to the US-APWR because it does not have reactor coolant system flow controllers.

15.3.3 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

15.3.3.1 Introduction

Other events that lead to a decrease in RCS flow rate are an instantaneous seizure of the rotor and a break of the shaft of an RCP. These events are treated separately in DCD Sections 15.3.3 and 15.3.4 but are combined and treated in this section of the SER. In these events, flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time.

15.3.3.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.3.3, summarized here as follows:

The application describes analyses that have been performed for events that could result in a decrease in reactor coolant system (RCS) flow rate, which can lead to an increase in the primary coolant temperature.

Analyses of the following events are described in this section:

- Reactor coolant pump rotor seizure
- Reactor coolant pump shaft break

Seizure of an RCP rotor and RCP shaft break events are PAs. As RCP rotor seizure event is designed to bound the response of an RCP shaft break, the system analysis, including radiological consequences, is reported in this section for the RCP rotor seizure transient only.

The instantaneous seizure of one RCP rotor during power operation would cause a rapid reduction in the reactor coolant flow (compared to the coastdown associated with an RCP trip) resulting in a decrease in core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure. This event is sometimes referred to as a locked pump rotor transient. Possible causes of a rotor seizure are bearing wear or bearing overheating due to loss of forced cooling or a coolant leak. However, the sudden stoppage of the RCP postulated in this scenario is more consistent with a failure affecting the rotating assembly, which results from a deformation that causes an interference with surrounding RCP components.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.3.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 15.3.3-15.3.4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," Revision 3, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can also be found in Subsection I, "Areas of Review," of Section 15.3.3-15.3.4 of NUREG-0800.

1. GDC 17, as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that design conditions of the RCPB are not exceeded and the core is cooled in the event of PAs.

2. GDC 27 and GDC 28, as they relate to the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained.

3. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

4. 10 CFR 100, as it relates to the calculated doses at the site boundary.

SRP acceptance criteria and also the technical rationale adequate to meet the above requirements are described in SRP Section15.3.3-15.3.4.

15.3.3.4 Technical Evaluation

The NRC staff reviewed the RCP rotor seizure and RCP shaft break event analyses, described in DCD Sections 15.3.3-15.3.4. The staff's review is performed in accordance with SRP Sections 15.3.3 –15.3.4. The acceptability of the system is based on meeting the requirements of GDC criteria and the SRP guidance described above. In DCD Sections 15.3.3-15.3.4, the applicant described the causes of the events, the event sequence, core and system performance, and radiological consequences due to these transient events. Also described in

the DCD are the analyses performed for these events, which include: evaluation models, input parameters and assumptions, and results of the analyses.

<u>Summary of technical information</u>

The particular events analyzed in this evaluation are the RCP rotor seizure and the RCP shaft break. During these events, flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage.

- RCP rotor seizure (locked rotor) event

  A reactor coolant pump rotor seizure (locked rotor) event is initiated by the instantaneous seizure of one RCP rotor during power operation. This postulated rotor seizure would cause a rapid reduction in the reactor coolant flow compared to the coastdown associated with an RCP trip, resulting in a decrease in core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure. This event is sometimes referred to as a locked pump rotor transient. Possible causes of a rotor seizure are bearing wear or bearing overheating due to loss of forced cooling or a coolant leak, or a deformation that causes an interference with surrounding RCP components. The seizure of an RCP rotor is a PA.

- RCP shaft break event

  A reactor coolant pump shaft break event is initiated by the instantaneous break (failure, or fracture and separation) of one of the reactor coolant pump shafts during power operation. This postulated shaft break would cause a reduction in the reactor coolant flow and decrease the core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure. Possible causes of this event are an undetected flaw in the shaft, or stresses caused by vibration or nonuniform temperatures. The break of an RCP shaft is a PA.

In DCD Subsection 15.3.3.3.1, the applicant described that in both cases the RCP failure causes a rapid decrease in flow in the affected loop. Reverse flow is then established in the affected loop, which becomes a core bypass path for some of the flow entering the downcomer from the intact loops. The loop reverse flow (and total core flow reduction) is greater after flow reversal for the shaft break case since the impeller is free to spin inside the pump casing. However, the initial abrupt flow decrease at the beginning of the locked rotor transient (before loop flow reversal occurs) results in lower core flow, because the RCP has higher resistance with the impeller locked. The limiting case for the locked rotor event is defined by assuming the RCP rotor is stopped prior to flow reversal, and that the pump resistance is changed to zero after the flow reverses in the affected loop. Therefore, the locked rotor design case also bounds the RCP shaft break. As a result, no analysis was performed or described for the shaft break event, which the staff finds acceptable based on the above justification.

In the subsections of DCD Section 15.3.3, for the RCP rotor seizure analysis, the applicant provided descriptions of the analysis for 1) core and system performance, 2) barrier performance, and 3) radiological consequences, each of which included: evaluation model, input parameters and initial conditions, and the results of the analyses. The details and results of these analyses are documented in the DCD Section 15.3.3 and its subsections, and the

results are depicted in the DCD tables and graphs. Also, the results of the analyses are further discussed in the staff evaluation below.

Staff Evaluation

Regulatory acceptance is based on meeting the requirements of GDCs 17, 27, 28, 31 and 10 CFR Part 100. The specific SRP criteria necessary to meet the requirements of these GDCs are provided in the regulatory basis section.

Also, the SRP specifies that the values of the parameters used in the analytical model must be suitably conservative. The RTDP is used to calculate the number of rods in DNB. The RTDP, described in WCAP-11397-P-A, "Revised Thermal Design Procedure," is an NRC approved methodology which suitably accounts for uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlations. The RTDP was not used to calculate the peak cladding temperature (PCT) nor the peak RCS pressure. The staff confirmed that those analyses used suitable conservative reactor power level, initial reactor operating parameters, and core heat transfer. The staff also confirmed that conservative scram characteristics were used, i.e., maximum time delay with the most reactive rod held out of the core.

- Conformance to GDC 17 relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function:

This GDC 17 criterion is interpreted by SRP Items 7 and 9 in SRP Acceptance Criteria, in SRP Section 15.3.3 – 15.3.4. Accordingly, LOOP should not be considered a single active failure for the RCP rotor seizure and shaft break events. Also, the acceptance criteria stated that the RCP rotor seizure event should be analyzed assuming a turbine trip in combination with LOOP and coastdown of the undamaged pumps.

The limiting single active component failure is the failure of one train of the RTS, because the RTS is the only mitigating system needed for this event. Any one of the remaining trains can provide the protection functions credited for the analysis. The low reactor coolant flow signal initiates the reactor trip. Additional details of the RTS are provided in DCD Section 7.2. The staff reviewed the applicant's limiting single active failure and the methodology for the selection of the limiting single failure, as described in the DCD, and determined that failure of one train of RTS as single active failure is acceptable.

In DCD Section 15.3.3.2, the applicant stated that the RCP rotor seizure results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could cause a LOOP and result in an RCP coastdown.

The applicant further stated that the RCP coastdown would not start until after the time of minimum DNBR; thus, the minimum DNBR for the entire transient is the same whether the offsite power is available or not available. Since the two cases (of offsite power) are equally limiting minimum DNBRs, the case where the offsite power is unavailable is not presented.

The staff requested further justification that this approach was bounding in RAI 306-2333, Question 15.3.1-1. In a response letter dated June 16, 2009, the applicant stated that the response to RAI 297-2287, Question 15.0.0-3, demonstrates the no-LOOP case bounds the case with LOOP assumed 3 seconds after the reactor/turbine trip with respect to DNBR, RCS pressure, main steam system pressure, and peak cladding temperature. As described in

Section 15.0.0.4 of this SER, the staff finds the response acceptable. Further, the staff finds RCP rotor seizure analysis meets the requirements of GDC 17 and SRP guidance in that LOOP is not considered as a single failure.

- Conformance to GDC 27 and GDC 28 relates to the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained.

The SRP Acceptance Criteria and SRP Technical Rationale provide guidance in meeting the GDC 27 and GDC 28 criteria. The specific criteria to meet the GDC requirements are: (1) the pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, and (2) fuel cladding integrity would be maintained by the minimum DNBR remaining above the 95/95 DNBR limit. If the DNBR falls below this value, fuel failure is assumed for all rods that do not meet the 95/95 DNBR limit.

The staff confirmed that the transients are simulated using the computer codes MARVEL-M and VIPRE-01M per the methodology described in MUAP-07010-P. NRC approval of MUAP-07010-P, including an evaluation of its restrictions and exceptions, and the applicable ranges of parameters when applied to this transient, is described in Section 15.0.2.4 of this SER.

DCD Section 15.3.3.3 provides details of the core and system performance. DCD Figures 15.3.3-1 "RCS Total and Loop Volumetric Flow versus Time, RCP Rotor Seizure – Rods in DNB Analysis" through 15.3.3-4 "RCS Average Temperature versus Time, RCP Rotor Seizure – Rods in DNB Analysis," depict the plots of system parameters (e.g., RCS total flow, reactor power, hot channel heat flux, and RCS average temperature) versus time for the rods in DNB analysis of the bounding RCP rotor-seizure/shaft-break transient with offsite power available. Also, the applicant provided DCD Figure 15.3.3-5, "Cladding Inside Temperature versus Time, RCP Rotor Seizure – Cladding Temperature Analysis," depicting the peak cladding temperature versus time. DCD Table 15.3.3-1, "Time Sequence of Events for RCP Rotor Seizure – Cladding Temperature Analysis," lists the key events and the times and DCD Table 15.3.3-3, "Summary of Results for RCP Rotor Seizure," summarizes the primary results of this analysis, including the peak local cladding temperature and oxidation fraction.

Also, DCD Section 15.3.3.4 provides the details of RCS barrier performance. For barrier performance evaluation for peak RCS pressure, the applicant used the same MARVEL-M model as in the core and system performance analysis. For DNB failure evaluation, the applicant used the VIPRE-01M code to calculate the minimum DNBR during the transient. NRC approval of MUAP-07010-P, which describes application of MARVEL-M and VIPRE-01M for this transient, is discussed in Section 15.0.2.4 of this SER.

Figures 15.3.3-6 "Reactor Power versus Time, RCP Rotor Seizure – RCS Pressure Analysis" through 15.3.3-10 "Steam Generator Pressure versus Time, RCP Rotor Seizure – RCS Pressure Analysis" depict the key system parameters (e.g., reactor power, core heat flux, RCP outlet pressure, and RCS average temperature) versus time for the peak RCS pressure evaluation of this analysis. The applicant stated that the RCP outlet pressure is the highest pressure in the RCS. According to Table 15.3.3-3 and Figure 15.3.3-8, "RCP Outlet Pressure versus Time, RCP Rotor Seizure – RCS Pressure Analysis" the RCP outlet pressure remains below 110 percent of the design pressure.

The staff reviewed the details of the analyses described in DCD Sections 15.3.3.3 and 15.3.3.4. The staff also reviewed the plots in the DCD Figures 15.3.3-1 through 15.3.3-10 which depicted

the key parameters versus timings of the RCP rotor seizure transient. The results of the analyses demonstrated that for a RCP rotor seizure, the number of rods predicted to be in DNB is less than 10 percent of the core, which is the value used in the radiological consequence analysis. Also, all rods not meeting the 95/95 limits are assumed to fail. The maximum fuel cladding temperature is about 1139°C (2082°F), which is below the limit of 1204°C (2200°F). The peak RCS and SG pressures remain below 110 percent of their design pressures.

However, since the applicant stated that the DNB analysis predicted a percentage of the fuel rods experience DNB, in RAI 353-2335, Question 15.3.3-1, the staff asked the applicant to provide the actual number of fuel rods that fail, and also to provide the minimum DNBR distribution for all rods in the core. The response confirmed that the calculated fraction of rods experiencing failure was less than the value used in the radiological consequences evaluation. The response also explained that the number of rods in DNB failure was determined from a hot channel factor ($F_{\Delta H}^{N}$) census curve. The DNB failure number was taken at the value of $F_{\Delta H}^{N}$, which just gives DNB occurrence, as determined from a VIPRE-01M sensitivity analysis. The SRP states that local flow conditions used in the core thermal-hydraulics model should consider the failed pump. Because the DCD did not identify any assumptions regarding the core inlet flow rate used in VIPRE-01M, the staff issued RAI 900-6313, Question 15.03.03-15.03.04-4. In order for the staff to find that the core design portion of this analysis was acceptable, the same RAI also asked the applicant to explain how the $F_{\Delta H}^{N}$ census curve was generated. Response to this question will be tracked as **Open Item 15.03-1**.

The applicant’s response to RAI 306-2333, Question 15.3.1-7, included a plot of the peak fuel centerline temperature as function of time for the 15.3.3 event. The staff finds this response acceptable because the plot demonstrated margin to the more limiting AOO fuel centerline safety limit described in Section 15.3.1.4 of this SER.

Based on the above discussions, the staff determined that the integrity of the reactor coolant and SG pressure boundaries are well maintained. Also, the staff finds the input parameters and assumptions are appropriately conservative and acceptable.

As discussed in Section 15.0.0.4 of this SER, the staff is satisfied that the Doppler coefficients selected for the 15.3 events were suitably conservative.

- Conformance to GDC 31, as it relates to the RCS being designed with sufficient margin to ensure the RCPB.

As discussed above, the RCP rotor seizure or shaft break will not result in an unacceptable stresses on the RCS and SG pressure boundary or on the ability to cool the reactor.
The peak RCS and SG pressures, and fuel clad temperature remain below the allowable values. There is adequate margin between the design/allowable values and accident analyses values. Therefore, based on the analyses of this event, the staff concluded that RCS is designed to meet the requirements of GDC 31, as it relates to maintain the integrity of the RCS pressure boundary to withstand this PA.

- Conformance to 10 CFR Part 100, as it relates to the calculated doses at the site boundary is demonstrated as described in Section 15.0.3.5 of this SER.

15.3.3.5 Combined License Information Items

There are no COL items listed for the RCP rotor seizure and rotor shaft break transients.

15.3.3.6 Conclusions

As a result of the open item, the staff is unable to finalize its conclusion on Section 15.3.3 in accordance with the requirements set forth in GDC 17, 27, 28, and 31 regarding the ability to insert control rods and to cool the core, and the 10 CFR 100 guidelines regarding radiological dose at the site boundary.

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

15.4.1.1 Introduction

An uncontrolled control assembly bank withdrawal from a subcritical or low power startup condition causes a positive reactivity insertion which increases reactor power, RCS temperature and RCS pressure. The event is classified as an AOO. The increase in reactor power and RCS temperature could result in violating the acceptable fuel design limits (SAFDLs), DNBR and fuel centerline temperature. The increase in RCS temperature will cause an accompanying increase in RCS pressure. Therefore, the maximum pressure is evaluated to ensure the RCS does not exceed 110 percent of the design value. The uncontrolled control bank movement could be caused by a malfunction of the reactor control system, or control rod drive system. Inherit design features of the control rod bank drive system, bank reactivity worth, and reactivity coefficients limit the power excursion.

The effects and consequences of an uncontrolled control rod assembly bank withdrawal from a subcritical or low power (e.g., startup-range) condition were analyzed to assure conformance with the requirements of GDCs 10, 17, 20, and 25. The review covers the description of the transient causes, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transient as compared with the acceptance criteria.

15.4.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.4.1, summarized here as follows:

The CRDMs are grouped into pre-selected bank configurations. The circuit design prevents the RCCA banks from being withdrawn in any manner other than their proper withdrawal sequence. Power supplied to the RCCA banks is controlled such that no more than two banks can be withdrawn at a time. The RCCA drive mechanisms are the magnetic latch type, and coil actuation sequencing provides variable speed travel. The maximum reactivity insertion rate is based on the simultaneous withdrawal of two sequential RCCA banks resulting in the maximum combined rod worth at maximum speed.

The neutron flux response to the continuous reactivity insertion due to RCCA movement is self-limiting. There is a rapid rise in power that is terminated by negative fuel temperature (Doppler) feedback immediately and then eventually by control rod insertion. The control rods are activated by an automatic trip signal that could emanate from the following conditions:

- High source range neutron flux
- High intermediate range neutron flux
- High power range neutron flux (low setpoint)
- High power range neutron flux (high setpoint)
- High power range neutron flux rate

The limiting single failure for these events is the failure of one train of the RTS. However, any one of the remaining trains is adequate to provide the protection functions credited in the analysis.

The evaluation model for these events uses TWINKLE-M and a one-dimensional (axial) model of the reactor to obtain power as a function of time. The use of a one-dimensional (1-D) TWINKLE model is claimed to lead to conservative results relative to a three-dimensional (3-D) calculation.

The TWINKLE-M results are used by VIPRE-01M to calculate DNBR and fuel temperature at the hot spot during the event. The TWINKLE-M code is documented in MUAP-07010, "Non-LOCA Methodology," [Reference 1], and the staff's SE is documented in **Reference 2 – Open Item 15.4-1**. The VIPRE-01M code is documented in MUAP-07009, "Thermal Design Methodology," [Reference 3], and the staff's SE is documented in **Reference 4 – Open Item 15.4-2.** The methodology includes the use of conservative initial conditions and other input parameters as shown in the following list:

1. Doppler feedback is conservatively estimated by multiplying the change in fast absorption cross section as a result of a change in calculated fuel effective temperature by a conservative multiplier (<1.0).

2. Moderator feedback is conservatively assumed to be based on the most positive moderator temperature coefficient during the reactor fuel cycle. This reduces the effect of the moderator in providing negative feedback after the power peak when heat has been conducted into the moderator.

3. Reactor power is assumed to be zero for purposes of determining temperature. This assumption leads to the use of a higher (and thus conservative) initial RCS temperature, which results in a greater fuel-to-water heat transfer coefficient, a greater fuel specific heat, and a smaller (less negative) Doppler feedback. This combination of effects

reduces the Doppler feedback beyond its value at the time in the fuel cycle when it is most conservative.

4. The initial values of reactor coolant average temperature and RCS pressure are assumed to be 2.2°C (4°F) above, and 0.21 MPa (30 psi) below the values corresponding to hot-standby conditions. This combination minimizes DNBR and maximizes fuel temperature calculated by VIPRE-01M.

5. The positive reactivity insertion rate of 75 pcm/s used in the analysis is greater than that for the simultaneous withdrawal of two sequential RCCA banks with the highest worth at the maximum speed 0.019 m/s (45 in/min). This takes into account uncertainties in RCCA reactivity worth.

6. Reactor trip simulation assumes conservative parameters for the trip reactivity, rod drop time, and trip delay time.

7. The most limiting axial and radial power shapes with the two highest combined worth banks is used for the DNBR and fuel temperature calculation.

8. The analysis assumes the initial power level for the kinetics calculation to be below that of any shutdown condition (10^{-13} of nominal power level). The low initial power, in conjunction with the maximum reactivity insertion rate, yields the highest peak heat flux.

9. For the peak RCS pressure evaluation a reactor coolant temperature of 4 °F and a pressurizer pressure 30 psi above the values the hot-standby conditions were used.

10. The radiological consequences of this accident are bounded by rod ejection which is described in DCD Section 15.4.8.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.4.1.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 15.4.1 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.4.1 of NUREG-0800.

The relevant requirements are:

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs.

2. GDC 13, which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 17, which requires provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.

5. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations and by satisfaction of any other SAFDL applicable to the particular reactor design.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this section.

5. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR Part 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

6. Performance of non-safety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY-77-439, SECY-94-084, and RG 1.206.

15.4.1.4 Technical Evaluation

The NRC staff reviewed the information described in US-APWR DCD Tier 1 and Tier 2 sections, in accordance with SRP Section 15.4.1. Because the US-APWR is a PWR, uncontrolled control rod assembly will be referred to as uncontrolled bank withdrawal for the remainder of this SER section. The acceptability of the system is based on meeting the requirements of GDC criteria and SRP acceptance guidance as described above in the regulatory basis section of this SE. The staff review focused on input and modeling conservatisms in TWINKLE-M and VIPRE-01M. The use of the 1-D TWINKLE code providing core power response to a sub-channel, thermal-hydraulics code such as VIPRE has been used in previously approved uncontrolled bank withdrawal analyses. The description of the TWINKLE-M and VIPRE-01M methodology used to evaluate uncontrolled bank withdrawal is described in Reference 1 and the staff's SE in Reference 2.

The following RAIs responses were evaluated by the staff to reach its safety conclusion.

RAI 308-2340, Question 15.4.1-1:

> TWINKLE-M is used to analyze the uncontrolled RCCA withdrawal event from zero power using a one-dimensional model. It is claimed that with the assumptions used, the methodology will lead to a conservative result. Show comparisons with the results of a three-dimensional model to justify that this approach is conservative.

In its response, dated July, 2009, the applicant provided [Reference 5] a comparison between the 1-D safety analysis model with two 3-D TWINKLE-M cases. The first 3-D case had the same reactivity insertion rate as the 1-D safety analysis case. The second 3-D TWINKLE-M case used a best estimate reactivity insertion rate by withdrawing the maximum bank worth from the hot zero power rod insertion limits. The applicant compared the peak core power, minimum DNBR and maximum fuel centerline temperature.

The results demonstrated that the 1-D TWINKLE-M safety analysis case was the most limiting. The relative ranking of the three cases turned out as expected, with a significant increase in DNBR and fuel centerline temperature margins using best estimate, maximum bank worths. As the applicant demonstrated that using the 1-D TWINKLE-M model is conservative for the safety analysis calculation the staff finds the response acceptable and **RAI 308-2340, Question 15.4.1-1 is resolved and closed**.

RAI 308-2340, Question 15.4.1-2:

> Explain what is meant by the statement "appropriate cross section data is selected to assure minimum Doppler feedback conditions" made in DCD Section 15.4.1.3.2 "Input Parameters and Initial Conditions."

In its response, dated July, 2009, the applicant explained [Reference 5] that the cross section data itself is not conservative but a multiplier less than 1 is used to decrease the Doppler temperature coefficient from the value given in Table 15.0-1, "Summary of Event Classification, Initial Conditions and Computer Codes." Decreasing the Doppler temperature coefficient will increase the transient core power rise yielding conservative DNBR, fuel centerline temperature and peak RCS pressure values. The applicant clarified the DCD statement and provided the multiplier value used to create a conservative Doppler temperature coefficient. The staff finds the applicant's response acceptable and **RAI 308-2340, Question 15.4.1-2 is resolved and closed**.

RAI 308-2340, Question 15.4.1-3:

> In DCD Section 15.4.1.3.2, explain why assuming "the effective multiplication factor to be one" maximizes the neutron flux peak?

In its response, dated July, 2009, the applicant explained [Reference 5] that starting at a subcritical condition the source range trip could occur and limit the power excursion. Starting a k_{eff} of 1.0 ensures that the core goes supercritical and hence undergoes a power excursion. In addition the applicant explained that a constant, conservative reactivity insertion is modeled until reactor trip which maximizes the power excursion. Using a constant conservative reactivity insertion until reactor trip is conservative; therefore the staff finds the applicant response acceptable and **RAI 308-2340, Question 15.4.1-3 is resolved and closed**.

RAI 308-2340, Question 15.4.1-4:

> How does the “conservative” withdrawal rate of 75 pcm/s for the uncontrolled RCCA bank withdrawal compare with the rate expected for both zero and full power initial conditions?

The applicant’s response, dated July, 2009, provided [Reference 5] a comparison between the maximum positive reactivity insertion rate used in the safety analysis calculation with a conservative calculation using the first core given in DCD Chapter 4.3, “Nuclear Design,” at BOC and EOC including a 10 percent margin for core design variations. Additional conservatism was added by multiplying the maximum differential rod worth by the maximum control rod speed. The applicant demonstrated significant margin between the conservatively calculated values and the assumed maximum positive reactivity insertion rate used in the DCD Chapter 15.4.1 analysis. The staff finds the DCD safety analysis method acceptable and **RAI 308-2340, Question 15.4.1-4 is resolved and closed**.

RAI 308-2340, Question 15.4.1-5:

> Are the most limiting axial and radial power shapes used for the bank withdrawal analysis calculated with ANC or TWINKLE-M? What is the control rod configuration that gives the most limiting shape?

In its response, dated July, 2009, the applicant explained [Reference 5] that the maximum positive reactivity insertion rates are found using a bottom-skewed power distribution. Therefore, a bottom-skewed power distribution is used in both TWINKLE-M and VIPRE-01M. The radial power distribution assumes a bounding $F_{\Delta H}^{N}$ value which is then confirmed on a reload basis using ANC. The bounding $F_{\Delta H}^{N}$ value is checked assuming control rods in overlap while the reactivity check is performed neglecting overlap to add conservatism. The worst reactivity addition presented was the simultaneous withdrawal of Banks-C and B which was well below the assumed bounding DCD safety analysis value. The staff agrees that a conservative reactivity insertion and radial power distribution are assumed but questions that a bottom-skewed VIPRE-01M axial power distribution yields the minimum DNBR. Therefore, RAI 889-6273, Question 15.4.1-10, was submitted requesting that the applicant justify the VIPRE-01M axial power distribution used to calculate DNBR – **This is being tracked as OI 15.4-3.**

RAI 308-2340, Question 15.4.1-6:

> The discussion of bank withdrawal from zero power event does not consider LOOP and neither does the dropped RCCA event, presumably because there is no reactor trip. How do these events take into account LOOP?

In its response, dated July 2009, the applicant noted [Reference 5] that at hot zero power that the turbine-generator is not connected to the grid. However, the applicant assumed a LOOP following the reactor trip. A comparison of plots of DNBR and fuel centerline temperature was provided in the response demonstrating no change in minimum DNBR and maximum fuel centerline temperature. This was to be expected as reactivity coefficients limit the transient. Therefore any actions which occur after the reactor trip will not have an impact on the minimum

DNBR or maximum fuel centerline temperature. The staff finds the response acceptable and **RAI 308-2340, Question 15.4.1-6 is resolved and closed**.

RAI 308-2340, Question 15.4.1-7:

> DCD Sections 15.4.1.5, "Radiological Consequences," and 15.4.2.5, also titled "Radiological Consequences," state that the radiological consequences for these AOOs are bounded by those calculated for a PA. It is understood that there are no radiological consequences for these events and this should be stated.

In its response, dated July 2009, the applicant stated that [Reference 5] the minimum DNBR stayed above, while the peak centerline and maximum RCS pressure, stayed below their respective acceptance criteria. As the fission product barriers remain intact there is no radiological consequence of this AOO. The staff agrees that the acceptance criteria are met and hence there is no radiological consequence. The staff finds the response acceptable and **RAI 308-2340, Question 15.4.1-7 is resolved and closed**.

Open Items

OI 15.4-1, Staff Safety Evaluation for Non-LOCA Topical Report, MUAP-07010, Revision 4.

OI 15.4-2, Staff Safety Evaluation for VIPRE-01M Topical Report, MUAP-07009, Revision 0.

OI 15.4-3, RAI 889-6273, Question 15.4.1-10 was submitted requesting that the applicant justify the VIPRE-01M axial power distribution used to calculate DNBR.

15.4.1.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD that pertain to this section.

15.4.1.6 Conclusions

As a result of the open items, the staff is unable to finalize its conclusion on Section 15.4.1 in accordance with the requirements of General Design Criteria 10, 13, 17, 20, and 25.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Introduction

An uncontrolled control rod bank withdrawal from power causes a positive reactivity insertion which increases reactor power, RCS temperature and RCS pressure. The event is classified as an AOO. The increase in reactor power and RCS temperature could result in violating the DNBR and fuel centerline temperature SAFDLs. The increase in RCS temperature will cause

an accompanying increase in RCS pressure. Therefore, the maximum pressure is evaluated to ensure the RCS does not exceed 110 percent of the design value. Without a manual or automatic reactor trip, the power mismatch and the rise of reactor coolant temperature could eventually result in DNB. To prevent damage to the fuel cladding, the RTS is designed to terminate the transient before the DNBR reaches the design limit. The uncontrolled control bank movement could be caused by a malfunction of the reactor control system or control rod drive system.

The effects and consequences of an uncontrolled control rod bank withdrawal from power were analyzed to assure conformance with the requirements of GDCs 10, 17, 20, and 25. The review covers the description of the causes of the transient, the transient itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transient as compared with the acceptance criteria.

15.4.2.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.4.2, summarized here as follows:

The control rod drive mechanisms are grouped into pre-selected bank configurations. The circuit design prevents the RCCA banks from being withdrawn in any manner other than their proper withdrawal sequence. Power supplied to the RCCA banks is controlled such that no more than two banks can be withdrawn at a time. The RCCA drive mechanisms are the magnetic latch type, and coil actuation sequencing provides variable speed travel.

A range of initial power levels, times in life, and reactivity insertion rates are run to determine the limiting conditions. Different initial reactor setpoints are examined, as different reactor trip setpoints will be reached depending on the inserted reactivity rate, time in life, and initial power level. The DNBR is the most limiting SAFDL for this event as peak rod powers increase less than 20 percent, which will not challenge the peak centerline fuel temperature. For slower reactivity insertion rates, the main steam safety valves open and pressurizer pressure drops before the high pressurizer pressure trip setpoint is reached, so that the over-temperature ΔT signal causes trip. At higher insertion rates, the power range neutron flux trip provides protection. At lower reactivity insertion rates from reduced power conditions, the over-temperature ΔT trip and high pressurizer pressure trip provide protection. The over-temperature ΔT trip has an axial offset penalty that reduces the setpoint if the axial power distribution is severe for DNB. The following reactor trip signals are assumed to be available to provide protection:

- High power range neutron flux (low setpoint)
- High power range neutron flux (high setpoint)
- High power range neutron flux rate
- Over-power ΔT
- Over-temperature ΔT
- High pressurizer pressure
- High pressurizer water level

The following automatic trip signals are assumed in safety analysis:

- High power range neutron flux (high setpoint)
- Over power ΔT
- Over temperature ΔT
- High pressurizer pressure

The event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a LOOP. This could cause an RCP trip, which affects DNBR. However, the minimum DNBR occurs before the LOOP so this case is not explicitly analyzed.

The limiting single failure for these events is the failure of one train of the RTS. However, any one of the remaining trains is adequate to provide the protection functions credited in the analysis.

MARVEL-M is a 1-D thermal-hydraulic, point kinetics model that is used to calculate the transient response of reactor power, reactor coolant pressure and temperature, hot spot heat flux, pressurizer water volume, and minimum DNBR. Minimum DNBR is calculated using VIPRE-01M based lookup tables. The MARVEL-M code is described in MUAP-07010, "Non-LOCA Methodology" [Reference 1], and the staff's SE is Reference 2. The following assumptions are used to assure a conservative DNBR:

1. Consistent with the use of the RTDP, the initial values of reactor power, and reactor coolant temperature, flow rate and pressure are assumed to be the nominal values without uncertainties.

2. Conservative axial and radial power distributions are assumed.

3. Trip reactivity is minimized for full power and for reduced power it is assumed to be that value that would result in a minimum shutdown margin at hot zero power corresponding to the most restrictive time in the core cycle.

4. Pressurizer spray is assumed to operate. This minimizes DNBR for any given combination of power and temperature.

5. The BOC case uses the minimum values for the magnitude of the moderator density coefficients and Doppler feedback coefficients whereas the EOC cases use the maximum values for feedback.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER

15.4.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 15.4.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.4.2 of NUREG-0800.

The relevant requirements are:

1. GDC 10, which requires that specified acceptable fuel design limits (SAFDLs) are not to be exceeded during normal operation, including the effects of AOOs.

2. GDC 13, which requires that the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 17, which requires provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.

5. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

Acceptance criteria adequate to meet the above requirements include:

6. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

7. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations and by satisfaction of any other SAFDL applicable to the particular reactor design.

8. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

9. The guidance in RG 1.105, "Instrument Spans and Setpoints," is used for its impact on the plant response to the type of AOOs addressed in this section.

10. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

Performance of non-safety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive

systems) must be evaluated and verified according to the guidance of SECY-77-439, SECY-94-084, and RG 1.206.

15.4.2.4 Technical Evaluation

The following RAIs responses were evaluated by the staff to reach its safety conclusion.

RAI 309-2345, Question 15.4.2-1:

> Discuss in detail the radial and axial power distributions used in the analysis and verify that the power peaking factors are at the design limits for a given power level.

The applicant stated in its response, dated May, 2009, [Reference 6] that the radial and axial power distributions are considered as part of the simplified DNBR lookup table. The VIPRE-01M runs, which created the MARVEL-M lookup table, use the power distributions given in DCD Subsections 4.4.4.3.1, "Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}{}^{N}$" and 4.4.4.3.2, "Axial Heat Flux Distributions." The $F_{\Delta H}{}^{N}$ value is set to maximum nominal HFP value with an adjustment for lower power conditions. DCD Section 4.4.4.3.2 describes the generic top-skewed axial power distribution used in Chapter 15 accidents that do not have extreme, localized axial shapes. The applicant provided both the radial and axial power distributions used in the VIPRE-01M calculations of the MARVEL-M lookup tables. The radial peaking factor (hot channel enthalpy rise) is set to the maximum, nominal HFP value and then adjusted for part power conditions. The generic axial power distribution is highly top peaked, which is conservative for DNBR calculations. Both of these power distributions assumptions are conservative; therefore, the staff finds the response acceptable and **RAI 309-2345, Question 15.4.2-1, is resolved and closed**.

RAI 309-2345, Question 15.4.2-2:

> Explain what is meant by a uniform radial power distribution resulting from RCCA bank withdrawal as stated in the last paragraph in DCD Section 15.4.2.3.3 "Results." Intuitively, one would expect radial power peaking within the assemblies from which the RCCAs are being withdrawn.

In its response, dated May, 2009, the applicant stated [Reference 6] that the uncontrolled control rod assembly withdrawal at power is characterized by the withdrawal of rods in a symmetric (or "uniform") pattern. The phrase "uniform radial power distribution," in the last paragraph of DCD Subsection 15.4.2.3.3, was used to explain that no significant skewing in the radial power distribution occurs. The applicant stated [Reference 6] the local power in the assemblies from which the RCCAs are being withdrawn could become larger than that before withdrawal; however, the relative power peaking is always within the design limit since the control rod positions assumed in this analysis are within the allowable range during normal operation. The staff agrees that the radial power distribution is relatively uniform and the control bank withdrawal patterns are grouped such that the radial power distribution is relatively uniform and will not violate the maximum nominal peaking factors adjusted for reactor power. As noted in the staff evaluation of RAI 309-2345, Question 15.4.2-1 the maximum, nominal radial peaking factors with an adjustment for lower power levels are used in the MARVEL-M DNBR lookup

table and hence conservative DNBR values will calculated. Therefore, the staff finds the response acceptable and **RAI 309-2345, Question 15.4.2-2, is resolved and closed**.

15.4.2.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD that pertain to this section.

15.4.2.6 Conclusions

The review of this event has considered possible initial conditions and the range of reactivity insertions, and the course of each resulting scenario, including the instrumentation response to the event. The methods used to determine the peak fuel rod response and reactor coolant boundary pressure have been reviewed, including the input into the analysis, such as power distributions, control rod worth and reactivity feedback from moderator and fuel temperature changes.

The staff conclusion takes into account the requirement of the GDC, specifically:

- GDC 10 and GDC 17, ensuring that the specified acceptable fuel design limits are not exceeded;

- GDC 13, ensuring that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges;

- GDC 20, ensuring that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded; and

- GDC 25, ensuring that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded.

These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and clad strain limits should not be exceeded), to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analyses have confirmed that the analytical methods and input data are conservative, and that SAFDLs will not be exceeded.

In all cases the minimum DNBR remains above the 95/95 limiting value. The lowest DNBR value occurs for a low reactivity insertion rate at BOC, HFP conditions with minimum reactivity feedbacks. Plots of peak rod power provided demonstrate a power increase less than 20 percent. A power increase this low will not challenge the peak centerline fuel temperature. Furthermore, plots were provided that demonstrate that RCS pressure remains below 110 percent of the design pressure.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 Introduction

The types of control rod misoperations that are assumed to occur include one or more rods moving or displaced from normal or allowed control bank positions such as dropped rods and rods left behind when inserting or withdrawing banks, or single rod withdrawal, and may include the automatic control system attempting to maintain full power.

15.4.3.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.4.3, summarized here, in part, as follows:

The application describes that control rod misoperation includes:

- one or more dropped rod cluster control assemblies (RCCAs) within a group or bank
- one or more misaligned RCCAs (relative to their bank)
- uncontrolled withdrawal of a single RCCA

The application identifies that a dropped or misaligned RCCA could be caused by failures or malfunctions of an RCCA drive mechanism or RCCA drive mechanism control equipment.

Movement of a single RCCA is never performed during normal operations. However, the capability to move a single RCCA exists in order to restore a dropped RCCA to its correct position under strict administrative procedural control. DCD Section 7.7.2.3 “Effects of Control System Failures” describes how no single equipment failure can cause an uncontrolled single RCCA withdrawal.

Therefore, the application identifies that single uncontrolled RCCA withdrawal is a PA and not an AOO. The applicant assigns an internal fuel failure criterion which is less than that typically allowed by other PAs. The RCCA misoperation transient evaluated in this section involves a single uncontrolled RCCA withdrawal outside of the dropped RCCA recovery procedure.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.4.3.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.4.3 of NUREG-0800 assuming that the events are AOOs and are summarized below. Review interfaces with other SRP sections can be found in Section 15.4.3 of NUREG-0800. The requirements, except for the uncontrolled single RCCA withdrawal, are:

1. GDC 10, which requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to assure that SAFDLs are not to be exceeded during any condition of normal operation, including the effects of AOOs.

2. GDC 13, which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 20, which requires, in part, that the protection system shall be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of AOOs.

4. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum DNBR remaining above the 95/95 DNBR limit for PWRs based on an acceptable correlation and by satisfaction of any other SAFDL applicable to the particular reactor design.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements in RG 1.105, “Instrument Spans and Setpoints,” are used for their impact on the plant response to the type of AOOs addressed in this section.

5. The most limiting plant system single failure, as defined in “Definitions and Explanations,” 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

6. Performance of non-safety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY-77-439, SECY-94-084, and RG 1.206.

For the uncontrolled withdrawal of a single RCCA, defined as a PA, the relevant acceptance criteria include the following:

1. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR falls below the limit, the fuel is assumed to fail.

2. The maximum radiological consequences shall be less than 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE).

3. A 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.

15.4.3.4 Technical Evaluation

One or more dropped RCCAs within a group or bank

This event is classified as an AOO. It causes a local power reduction, which in turn causes an operational disturbance of the reactor, namely, a decrease in reactor power and/or a decrease in average reactor coolant temperature. Although there are a number of direct and indirect means of detecting the dropped RCCA(s) (the assumption is made that they are not detected. In this case), other RCCAs could be withdrawn to compensate for the reactivity decrease in order to restore power and/or average coolant temperature to match the turbine demand. The dropped rod event is therefore analyzed assuming automatic rod control. If RCCAs were withdrawn, the reactor power would be restored and the concern is the increase in hot channel heat flux. If the control rod control system were in manual control and no operator action were taken, the RCS temperature and pressure would decrease until a new equilibrium condition was reached or the reactor was tripped automatically on low pressurizer pressure.

The event is analyzed using MARVEL-M to determine the transient response. It provides power and coolant pressure and temperature and uses the RTDP to determine DNBR. In order to provide a bounding nuclear enthalpy rise hot channel factor ($F_{\Delta H}{}^{N}$) which is used to evaluate the hot spot heat flux, and hence determines the limiting DNBR, various combinations of dropped RCCA locations and rod worths are identified and modeled using ANC.

The analysis uses the following assumptions:

1. Consistent with use of the RTDP, the initial values of reactor power, and reactor coolant average temperature and pressure, are assumed to be the nominal full power values without uncertainties.

2. The moderator density coefficient is assumed to have the minimum value and the Doppler power coefficient is assumed to have its minimum value. This maximizes heat flux in the initial stage of the transient (i.e., the power overshoot phase).

3. Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, and signal processing delay) are used.

   The inserted reactivity for the dropped RCCA is 0.25% Δk/k, which is greater than the maximum reactivity insertion resulting from one RCCA dropping from fully withdrawn to fully inserted at rated power. It is inserted instantaneously.

4. The rod control system is assumed to be in the automatic control mode. This assumption results in rod withdrawal to match turbine load, which maximizes reactor power and core heat flux. Due to this assumption, no reactor trip occurs.

5. Pressurizer heaters are assumed to be off. This minimizes RCS pressure.

6. A bounding constant value of $F_{\Delta H}^{N}$ = 1.90 is used based on the analysis of different configurations.

The results of this event are shown in the DCD by plotting reactivity, reactor power, hot spot heat flux, RCS pressure and average temperature, and minimum DNBR vs. time. As no reactor trip is predicted, the reactor reaches a new equilibrium after the transient. The minimum DNBR remains well above the limiting value. Hence, no fuel centerline melting is expected. The core pressure is also not strongly impacted and will not exceed 110 percent of design pressure.

<u>RAIs for one or more dropped RCCAs</u>

The following RAIs were submitted concerning various aspects of the parameters, assumptions, and results of the safety analyses presented on this aspect of DCD Section 15.4.3.

RAI 310-2346, Question 15.4.3-2:

> Provide a list of the various combinations of dropped RCCA locations and rod worths that are used to identify the limiting hot channel factor for that event.

The applicant responded [Reference 7] to RAI 15.4.3-2 by providing a plot showing many combinations of multiple dropped RCCAs within a group at BOC, MOC and EOC. The sample cases used best estimate dropped rod worths and the corresponding $F_{\Delta H}^{N}$ values plus uncertainty. The plot demonstrated that the DCD case had a lower (more conservative) DNBR. The applicant's response is acceptable and RAI 310-2346, Question 15.4.3-2 is closed.

RAI 310-2346, Question 15.4.3-3:

> How does the assumed dropped rod worth of 0.25 percent Δk/k compare with the actual maximum dropped rod worth?

The applicant provided [Reference 7] a table of best estimate plus uncertainty, maximum dropped rod worths at BOC, MOC and EOC from the ARO to ARI position. The best estimate plus uncertainty values demonstrated that there was sufficient margin to the assumed dropped rod worth of 0.25 percent Δk/k. Therefore, the applicant demonstrated the assumed 0.25 percent Δk/k is bounding and RAI 310-2346, Question 15.4.3-3 is closed.

RAI 310-2346, Question 15.4.3-4:

> How does the assumed hot channel factor of 1.90 compare to the maximum value expected during a dropped rod event.

The applicant provided [Reference 7] a table of best estimate plus uncertainty $F_{\Delta H}^{N}$ values at BOC, MOC and EOC for a number of dropped rods. The best estimate plus uncertainty values demonstrated that there was sufficient margin to the assumed 1.90 $F_{\Delta H}^{N}$ value. Thus, the applicant demonstrated the assumed 1.90 $F_{\Delta H}^{N}$ is bounding and RAI 310-2346, Question 15.4.3-4 is closed.

One or more misaligned RCCAs (relative to their bank)

RCCA misalignment is considered an AOO. If it should happen due to a malfunction, there are direct and indirect ways in which the misalignment can be detected so that the situation can be rectified. The assumption is made that it is not detected and the issue is whether the modified core power distribution differs from the design power distribution so as to reduce margin to the fuel design limits.

No transient analysis is needed for this event. Various static rod misalignment scenarios (single rods or groups within banks) are identified and modeled with ANC to calculate a new bounding $F_{\Delta H}^{N}$ for use in the DNBR calculation. Scenarios considered include a single RCCA fully inserted, one RCCA fully withdrawn with the remaining bank of RCCAs at their insertion limits and other intermediate misalignment conditions within +/- 24 steps.

The DNBR analysis is done with steady-state VIPRE-01M calculations using the RTDP. Nominal conditions are used for the initial conditions consistent with the RTDP approach. The minimum DNBR is above the acceptance criterion and the linear heat generation rate is low enough so that no fuel centerline melting would occur. In addition, this mild transient will not strongly affect system pressure and the limit of 110 percent of design pressure will not be exceeded.

RAIs for one or more misaligned RCCAs

Two requests for additional information were submitted concerning various aspects of the parameters, assumptions, and results of the safety analyses presented on this aspect of DCD Section 15.4.3.

RAI 310-2346, Question 15.4.3-5:

> Calculations are carried out to determine the limiting configuration with one or more misaligned RCCAs. What configurations were sampled? It is assumed that the limiting misalignment is with one RCCA completely withdrawn. What is the effect of two RCCAs, or a control rod group, withdrawn?

In the response [Reference 7] the applicant states, “It is MHI’s position that to misalign more than one RCCA within a group or one RCCA in more than one group would require multiple failures and is therefore not an AOO but a PA.” The applicant did provide sample cases with one to three Control Bank D RCCAs misaligned and compared the $F_{\Delta H}^{N}$ values to one RCCA fully inserted and withdrawn cases. For the sample cases provided, one fully withdrawn and inserted case bound the Control Bank D misalignment $F_{\Delta H}^{N}$ values but the differences were very small. As noted in the applicant’s response, the limiting configuration for control rod misalignment may be cycle specific. Therefore, the staff requires a commitment from the applicant that this event must be analyzed, and acceptance criteria of no predicted fuel failures must be met, on a cycle-specific basis. Also, the staff could not determine if the applicant evaluated other possible RCCA misalignment configurations, hence **RAI 888-6274, Question 15.4.3-12 was written as a follow-on to RAI 310-2346, Question 15.4.3-5; RAI 888-6274 is OI 15.4-4.**

RAI 310-2346, Question 15.4.3-6:

> It is stated that the minimum DNBR calculated for the misaligned RCCA satisfies the acceptance criterion. What is the calculated value for the minimum DNBR?

Quantitative results were provided for DNBR in response to RAI 15.4.3-6 [Reference 7]. The minimum DNBR was greater than the value assumed as the safety analysis limit. The response of the applicant is acceptable and RAI 310-2346, Question 15.4.3-6 is closed.

Uncontrolled withdrawal of a single RCCA

This event requires multiple operator or system failures and, therefore, MHI considers the event as a PA.

The event leads to a core power increase similar to a bank withdrawal at power with a concurrent adverse change in power distribution that can exceed the design power distribution. The following automatic trip signals are assumed to be available to provide protection:

- high power range neutron flux rate
- high intermediate range neutron flux
- high power range neutron flux
- low pressurizer pressure
- over-power ΔT
- over-temperature ΔT

The limiting single failure for this event is the failure of one train of the RTS. However, any one of the remaining trains is adequate to provide protection. The event is not explicitly modeled with MARVEL-M. Instead, the transient results from the analysis of the limiting bank withdrawal event is used along with a bounding $F_{\Delta H}^{N}$ characteristic of a single RCCA withdrawal. The bounding value is obtained by considering various combinations of single RCCA location and rod worths and the steady-state neutronics code ANC. The result is the use of $F_{\Delta H}^{N}$ = 1.90 for the event. VIPRE-01M is then used to determine the minimum DNBR using the RTDP.

The changes in core parameters, such as power and RCS temperature and pressure, are assumed to be bounded by those for the transient associated with bank withdrawal. The resulting analysis shows that the fraction of fuel rods predicted to be in DNB is less than 5 percent. Therefore, the radiological consequences of this accident are bounded by rod ejection which is described in Section 15.4.8 of this SER.

RAIs for uncontrolled withdrawal of an RCCA

Five requests for additional information were submitted concerning various aspects of the parameters, assumptions, and results of the safety analyses presented on this aspect of DCD Section 15.4.3.

RAI 310-2346, Question 15.4.3-1:

> If the withdrawal of a single RCCA is not an AOO, it should be classified as a PA. Provide a probabilistic analysis to justify that it should be considered a PA. Provide explicit details and an analysis justifying the reclassification of the uncontrolled single

RCCA withdrawal event. Specifically, the staff is requesting risk assessment studies and radiological consequences.

In response to RAI 15.4.3-1 the applicant provided [Reference 7] a system description, failure modes and effects analysis, fault tree analysis and final probabilistic analysis to justify the classification of the event. The failure modes and effects analysis determined that, when a single failure in the CRDMCS occurs, the control rods are either dropped or are inoperable. The probabilistic risk assessment calculated a frequency of an uncontrolled single RCCA withdrawal approximately one order of magnitude less than a Small Break LOCA.

The single movement of an RCCA can also be caused by multiple operator errors. As stated in the DCD Section 15.4.3.1, "Identification and Causes and Frequency Classification," the likelihood of this error is very low. Movement of a single RCCA is not procedurally allowed under normal operations. The capability does exist to move a single RCCA but only under strict procedural control to recover a dropped RCCA.

Historically, the uncontrolled RCCA withdrawal was classified as a Condition III infrequent event as defined in ANSI N18.2. Per SRP 15.4.3 the staff can determine if a single RCCA withdrawal control rod can be classified as an AOO or PA depending on the reactor control rod design. The staff agrees that a single RCCA withdrawal can be considered a PA accident based on the fact that multiple failures would have to occur for the US-APWR control rod system to fail.

RAI 310-2346, Question 15.4.3-7:

What are the configurations sampled to determine the limiting condition for the uncontrolled withdrawal of an RCCA?

The applicant responded that one RCCA is fully withdrawn with the other RCCAs in the control group at the power dependent insertion limits. All RCCAs in the group were evaluated. The bounding $F_{\Delta H}^{N}$ of 1.90 was used to evaluate the DNBR. **The applicant provided the necessary information for the staff to understand the methodology but did not provide information if partial power cases were evaluated. Therefore, this RAI remains an OI – Open Item 15.4-5.**

RAI 310-2346, Question 15.4.3-8:

For the withdrawal of a single RCCA, it is understood that the minimum DNBR at the hot spot will not satisfy the 95/95 limits. How is the number of rods below the DNBR limit obtained?

The applicant stated [Ref 7] that a sensitivity analysis of the heat flux due to distorted radial power distribution is done and a search is performed for the $F_{\Delta H}^{N}$ that just gives DNB. The other VIPRE-01M boundary conditions are obtained from the control rod bank withdrawal described in DCD Section 15.4.2. The number of rods in DNB is obtained by counting the number of rods that have $F_{\Delta H}^{N}$ values greater than or equal to the value that just gives DNB. The staff agrees that the methodology described will conservatively predict the number of rods in DNB. Therefore, the response is acceptable and RAI 310-2346, Question 15.4.3-8, is closed.

RAI 310-2346, Question 15.4.3-9:

> What is the fuel centerline temperature for the withdrawal of a single RCCA?

To demonstrate that the fuel centerline temperature remained below the melting point the applicant used a more detailed analysis method [Reference 7]. Conservative input values of initial core power and hot channel factor were used. The fuel centerline temperature uncertainty was added to the initial fuel temperature. Parametric cases were then run to find the reactivity insertion rate yielding the highest centerline temperature. The analysis demonstrated that the highest calculated temperature was less than the minimum fuel pellet melting temperature. The RAI response did not provide any details of the methods used to arrive at this conclusion. **As such it is impossible for the staff to reach a safety conclusion regarding fuel centerline melt. Therefore, RAI 310-2346, Question 15.4.3-9 remains an OI – Open Item 15.4-6.**

RAI 310-2346, Question 15.4.3-10:

> Specify which steady-state core design codes were used throughout the analysis and include references to the codes. Be specific with code versions and provide reference. [Note that this RAI is relevant to all events analyzed in Section 15.4.3].

The applicant responded [Reference 7] the ANC is the code used and the references are identified in Section 4.3 “Nuclear Design” of the DCD. The response is acceptable and RAI 310-2346, Question 15.4.3-10, is closed.

Open Items

OI 15.4-4. RAI 888-6274, Question 15.4.3-12 was written as a follow-on to RAI 310-2346, Question 15.4.3-5, to determine if other possible RCCA misalignments have been considered.

OI 15.4-5. RAI 904-6324, Question 15.4.3-13. To determine if partial power RCCA withdrawal cases were evaluated.

OI 15.4-6. RAI 310-2346, Question 15.4.3-9, remains open as the applicant did not provide staff enough information to determine if fuel centerline temperature was calculated conservatively.

15.4.3.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD that pertain to this section.

15.4.3.6 Conclusions

As a result of the open items, the staff is unable to finalize its conclusion on Section 15.4.3 in accordance with the requirements of General Design Criteria 10, 13, 20, and 25.

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature

15.4.4.1 Introduction

The startup of an inactive loop in a PWR may cause either increased core flow or introduction of cooler or de-borated water into the core. These AOOs result in an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. This section is intended to be applicable to all such AOOs.

15.4.4.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has not provided a DCD Tier 2 description in Section 15.4.4 for the following reason:

This section is not applicable to the US-APWR, because power operation with an inactive loop is not allowed by the Technical Specifications.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.4.4.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.4.4 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.4.4 of NUREG-0800.

The relevant requirements are:

1. GDC 10 and GDC 20, as they relate to the reactor coolant system (RCS) being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations and AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15 and GDC 28, as they relate 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 and AOOs.

4. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during AOOs. This is accomplished by ensuring that appropriate margins for malfunctions, such as stuck rods, is accounted for.

5. The basic objectives of the review of the AOOs described above are:
    - to identify which of the AOOs are the most limiting.
    - to verify that, for the most limiting AOOs, the plant responds in such a way that the criteria regarding fuel damage and system pressure are satisfied.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations (see DCD Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this section.

5. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

6. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY-77-439, SECY-94-084, and RG 1.206.

15.4.4.4 Technical Evaluation

This event is not analyzed because power operation with an inactive loop is not allowed by Technical Specifications. Limiting Condition of Operation (LCO) 3.4.4 states that four reactor coolant system loops shall be operable and in operation in Modes 1 and 2. If this is not the case, the required action is to bring the reactor to Mode 3, which is hot standby. Therefore, the Technical Specifications preclude consideration of this event.

In Mode 3 and below less than four RCPs can be in operation (see LCO 3.4.5). The application does not address lower mode startup of an inactive loop. Per SRP 15.4.4, Rev. 2, “An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two or three loops initially operating) or the effects referenced to a limiting case.”

Open Items

OI 15.4-7. RAI 903-6325, Question 15.04.04-15.04.05-1. Applicant did not address Startup of an Inactive Loop for operating Modes 3-5.

15.4.4.5 Combined License Information Items

There are no COL information items identified for this section in Table 1.8-2 of the DCD:

15.4.4.6 Conclusions

This event is not analyzed because power operation with an inactive loop is not allowed by Technical Specifications. Limiting Condition of Operation (LCO) 3.4.4 states that four reactor coolant system loops shall be operable and in operation in Modes 1 and 2. If this is not the case, the required action is to bring the reactor to Mode 3, which is hot standby. Therefore, the Technical Specifications preclude consideration of this event.

Lower modes of operation (i.e., Mode 3 and below) are not presented in the DCD. Therefore, the staff cannot make a safety finding on startup of an inactive loop at lower mode conditions. **This is Open Item 15.4-7.**

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

The US-APWR is not a BWR (it is a PWR) and, thus, this section is not applicable.

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

15.4.6.1 Introduction

The boron dilution event is the result of a malfunction of the chemical and volume control system (CVCS) that causes the inadvertent addition of water with low boron concentration into the RCS and the failure of the operator to respond to indicators. This results in a positive reactivity addition to the core. The event is considered an anticipated operational occurrence (AOO) and needs to be considered for each mode of operation.

15.4.6.2 Summary of Application

DCD Tier 1: There were no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant provided a DCD Tier 2 description in Section 15.4.6, summarized here as follows:

An inadvertent decrease in boron concentration in the reactor coolant is classified as an AOO. In addition to the general AOO acceptance criteria described in DCD Section 15.0.0.1.1, SRP 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)" imposes additional guidance for the minimum time intervals to be available for operator actions between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost (criticality).

The inadvertent decrease in reactor coolant boron concentration (i.e., boron dilution) event is evaluated during all modes of operation including refueling conditions, shutdown conditions, the

beginning of reactor startup, and power operation. DCD Table 15.4.6-1, “Summary of Analysis Input Parameters and Results of Boron Dilution Analysis,” provides the operating parameters and conditions associated with each mode of operation for the boron dilution event for the US-APWR.

Boron dilutions during refueling (Mode 6) or during shutdown operation with no reactor coolant pumps (RCPs) running (Modes 4 & 5) are not analyzed based on TS LCOs 3.4.6, 3.4.7 and 3.4.8 and their associated actions.

DCD Section 15.4.6.3, “Input Parameters and Initial Conditions,” discusses the core and system performance including discussion of the evaluation model, input parameters and initial conditions, and the calculation results. The acceptance criteria for these calculations is that the minimum calculated time available for operator action is greater than the minimum allowed time interval described in the acceptance criteria for Chapter 15.4.6 of the SRP.

For cases where reactor power does not increase during the transient, barrier performance is bounded by the results of the inadvertent CVCS operation event documented in DCD Tier 2 Section 15.5, “Increase in Reactor Coolant Inventory.” For cases where the transient is initiated at power and reactor power increases, barrier performance is bounded by the results for the uncontrolled control rod assembly bank withdrawal at power event documented in DCD Tier 2 Section 15.4.2, “Uncontrolled Control Rod Assembly Withdrawal at Power.”

For transients initiated at power, the minimum DNBR remains above the 95/95 limit so that fuel failure is not predicted. As reactor coolant system pressure remains well below 110 percent of its system design pressure for all cases, the integrity of the RCPB is maintained. For all cases with the reactor shut down (or tripped), sufficient indications are available to alert the operator to the uncontrolled reactivity addition and sufficient time is available for the operators to diagnose the situation and take corrective action before criticality or post-trip return to criticality occurs.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.4.6.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in NUREG-0800, Section 15.4.6, “Inadvertent Decrease in Boron Concentration in the Reactor Coolant System,” and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.6.

1. General Design Criterion (GDC) 10, as it relates to the reactor core and its coolant, control, and protection systems with appropriate design margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

2. GDC 13 as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to the RCS and its auxiliary, control, and protection systems with sufficient design margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

4. GDC 26, as it relates to the capability of control rods to reliably control reactivity changes to assure that, under conditions of normal operation, including anticipated operational occurrences (AOOs), and with appropriate margin for malfunctions like stuck rods, specified acceptable fuel design limits are not exceeded.

5. The general objective of the review of moderator dilution events is to confirm either of the following conditions is met:

   A. The consequences of these events are less severe than those of another transient that results in an uncontrolled increase in reactivity and has the same anticipated frequency classification.

   B. The plant responds to events such that the criteria regarding fuel damage and system pressure are met and the dilution transient is terminated before the shutdown margin is eliminated.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained so that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations with SRP Section 4.4.

3. An incident of moderate frequency should not generate a more serious than moderate plant condition without other faults occurring independently.

4. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:

   A. During refueling: 30 minutes.

   B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.

5. The applicant's analysis of moderator dilution events should use an acceptable analytical model. Staff must evaluate any proposed unreviewed analytical methods. The reviewer initiates an evaluation of new generic methods. The following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown and cold shutdown. Parameters and assumptions in the analytical model should be suitably conservative. The following values and assumptions are acceptable:

A. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent to account for power-measurement uncertainty. The analysis may use a smaller power-measurement uncertainty if justified adequately.

B. The boron dilution is assumed to occur at the maximum possible rate.

C. Core burnup and corresponding boron concentration must yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. The core burnup must be justified by either analysis or evaluation.

D. All fuel assemblies are installed in the core.

E. A conservatively low value is assumed for the reactor coolant volume.

F. For analyses during refueling, all control rods are withdrawn from the core. As an alternate assumption requires adequate justification and delineation of necessary controls, the alternate assumption remains valid.

G. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1 percent) is assumed prior to boron dilution.

H. A conservatively high reactivity addition rate is assumed for each analyzed event to take into account the effect of increasing boron worth with dilution.

I. Conservative scram characteristics are assumed (i.e., maximum time delay with the most reactive rod out of the core).

15.4.6.4 Technical Evaluation

If the reactor is at power (Mode 1) under automatic rod control when the dilution begins, rods will be automatically inserted into the core in order to maintain the programmed RCS temperature. Unanticipated rod stepping can be recognized in the control room and confirmed by observing the rod position indications and, in the limiting case, an alarm will actuate when the rods reach their respective RIL. If the unplanned dilution continues after this alarm, the reactor is tripped and could return to criticality under HZP conditions if no operator action is taken. The operator action time is the time between the RIL annunciator alarm and return to criticality.

If the reactor is at power (Mode 1) under manual rod control when the dilution begins, reactor power and RCS average temperature will increase. The resulting transient is similar to the uncontrolled RCCA bank withdrawal at power transient and results in a reactor trip if no action is taken to mitigate the dilution. If the unplanned dilution continues after the reactor is tripped, the reactor may return to criticality under HZP conditions if no operator action is taken. When the control rods are under manual control, which can occur during all modes, the operator action time is from the start of the dilution and the return to criticality.

If the reactor is in a normal startup operation (Mode 2), reactivity is added by manual (planned) dilution or manual control rod withdrawal. An inadvertent dilution during this operation could

result in a power escalation and high source range neutron flux reactor trip before the operator manually blocks the source range reactor trip.

If the reactor is in hot standby (Mode 3) or shutdown (Modes 4 and 5) with RCPs running, the analysis assumes the transient starts with all rods fully inserted (except the highest reactivity-worth rod fully withdrawn) and the initial RCS boron concentration is calculated to ensure the operators have at least 15 minutes to criticality from the point when the high source range neutron flux alarms and criticality is reached.

As explained in DCD Section 15.4.6.2, the boron dilution event is considered for each mode of operation. However, the applicant explained, in response to RAI 682-5367, Question 15.4.6-6, that the potential for boron dilution during refueling (Mode 6) or during shutdown operation with no RCPs running (Modes 4 and 5) is extremely low, based on TS LCOs 3.4.6, 3.4.7 and 3.4.8, which place strict operating requirements in those modes. Consequently, no quantitative boron dilution analysis was performed for Mode 6 and Modes 4 and 5 with no RCPs running. The staff's review of the TS wording notes that "planned dilution" appears in the revised TS Conditions. It was the staff's understanding that no dilution would occur in Modes 4 and 5, with no RCPs running and Mode 6. Also, it is not clear to the staff how an RCS dilution can occur with all unborated water sources isolated. Therefore, **Open Item 15.4-8** has been created to resolve the staff's concern.

Analysis Approach and Assumptions

Modes 1 and 2

The initial boron concentration, C1, in Table 15.4.6-1, "Summary of Analysis Input Parameters and Results of Boron Dilution Analysis" defines the lowest, undiluted, critical boron concentration with the control rods at the RIL. The boron concentration, C2, in Table 15.4.6-1 is the HZP, critical boron concentration with the control rods at the RILs. The critical boron concentrations are calculated using the ANC code for which approval is given in the DCD, Chapter 4.3. Using the boron and water mass equilibrium equations, a single equation can be derived which relates the time to reach boron concentration C2 when starting at C1.

Modes 3-5, one RCP running

The initial boron concentration, C1, in Table 15.4.6-1, is the boron concentration that yields an operator action time of at least 15 minutes from the high source range neutron flux alarm. The boron concentration, C2, is the critical boron concentration at each mode assuming all-rods-in (ARI) with the highest worth rod fully withdrawn. A simplified, subcritical multiplication equation and setting the high source range neutron flux alarm analytical limit 0.8 decades above the background count rate determines the C1 boron concentration.

The following conservative inputs are used to determine the minimum operator action times:

- Highest core reactivity condition, BOC, zero Xenon concentration, is evaluated
- A maximum charging flow rate of 265 gpm, with a 15 gpm uncertainty, based on the isolation valve high flow rate setpoint
- Minimum RCS volume
- Maximum dilution flow density at 32 ^{o}F and 14.7 psia
- 100 ppm uncertainty added to critical boron concentrations
- Zero ppm boron dilution water

- Conservative differential boron worths

Modes 4 and 5 no RCPs running and Mode 6

No boron dilution times were calculated for Modes 4 and 5, with no RCPs running, and Mode 6. No dilution times are calculated based on isolating all dilution sources in these conditions based on TS LCOs 3.4.6, 3.4.7 and 3.4.8 and their associated actions.

<u>Credited Alarms</u>

DCD Table 15.4.6-1:

- Mode 1 - RIL alarm signals start of event (rods in automatic mode)

- Mode 1 - High power range neutron flux trip, or OTΔT trip, signals start of event (rods in manual mode)

- Mode 2 – High source range neutron flux trip, high power range neutron flux trip, or OTΔT trip signals start of event (rods in manual mode)

- Mode 3 – Reactor makeup water, low rate deviation alarm, the boric acid flow rate deviation alarm, and the high primary makeup water flow rate alarm.

- Mode 4 - Reactor makeup water, low rate deviation alarm, the boric acid flow rate deviation alarm, and the high primary makeup water flow rate alarm.

- Mode 5 - Reactor makeup water, low rate deviation alarm, the boric acid flow rate deviation alarm, and the high primary makeup water flow rate alarm.

In RAI No. 311-2347, Question 15.4.6-5, the staff asked the applicant for more details regarding the calculations done to determine the time available for operator action during the course of a boron dilution event since this was not discussed in the DCD. Specifically, the staff asked the applicant to provide the boron and water mass equilibrium equations used in the calculations. In the response to Question 15.4.6-5 [Reference 8], the applicant provided the boron mass equilibrium equation and the water mass equilibrium equation with a description of the corresponding input parameters and initial condition assumptions. Combining the two equations and assuming that the boron concentration being introduced to the RCS is zero ppm, the resultant ordinary differential equation can be solved analytically for the boron concentration as a function of time. The input parameters, assumed to be constant, in the equation are the RCS volume, the dilution flow rate, the RCS coolant density, the incoming dilution coolant density, and the initial boron concentration before the dilution occurs. For the at-power cases, this equation can then be solved for the time between the initial critical boron concentration (C1) and the final HZP, critical boron concentration (C2).

For the hot standby and shutdown (Modes 3-5) cases, with at least one RCP running, the initial boron condition is mode-specific and is determined by ensuring that the TS required SDM is maintained. The final, or critical, boron concentration, C2, is based on the assumption that the reactor is critical with all rods fully inserted except for the most reactive single rod, which is assumed to be fully withdrawn from the core. The standby and shutdown case dilution times in DCD Table 15.4.6-1 are calculated by accounting for the fact that the dilution time is defined

with respect to the high source range neutron flux alarm activation in accordance with the applicant's variable shutdown margin (V-SDM) methodology. This requires a simple modification to the dilution time equation by introducing the boron concentration that will trigger the high source neutron flux alarm, hereafter referred to as the alarm concentration. The applicant shows how the critical concentration can be related to the alarm concentration by providing two more equations that describe subcritical multiplying systems. The first describes the "analytical limit" of the high source range detector, in terms of decades above the background radiation source, and is related to the rate of subcritical multiplication. The second equation is a generic linear relationship for the boron concentration as a function of k-effective with the slope and y-intercept being arbitrary constants. With this generic linear equation, three specific forms can be written for the initial boron concentration, the final, or critical, concentration, and the alarm concentration. Combining the four equations describing subcritical multiplication, a single equation relating the initial boron concentration, the final concentration, the alarm concentration, and the analytical limit of the source range detector is formed. This resultant equation, which is solved for the alarm concentration, can finally be combined with the original dilution time equation, which has the same form but uses the alarm concentration in place of the initial, critical boron concentration. Based on the description given by the applicant regarding the calculation for the dilution time for the at-power, standby, and shutdown cases, and the staff's review of the input parameters and underlying assumptions, the staff believes that the applicant has appropriately calculated the dilution time for all cases, and Question 15.4.6-5 is therefore closed. However, in RAI 708-5455, Question 15.4.6-9, the staff requested additional justification for using the simplified, subcritical multiplication equation to determine the assumed increase in neutron population 0.8 decades above background.

In RAI 708-5455, Question 15.4.6-9, the staff pointed out that the simplified subcritical multiplication equation is non-conservative when used to predict the increase in neutron population during a continuous dilution. Therefore, crediting this alarm and the associated operator response time would also be non-conservative. The applicant responded [Reference 9] that the US-APWR design has other indicators of a Modes 3-5 boron dilution available. The applicant agreed to modify DCD Table 15.4.6-1, "Summary of Analysis Input Parameters and Results Boron Dilution Analysis" to credit one more of the following alarms: reactor makeup water, low rate deviation alarm, the boric acid flow rate deviation alarm, and the high primary makeup water flow rate alarm. This is **Confirmatory Item 15.4.6-a**.

Therefore, the initial, undiluted boron concentration, C1, is still based on the simplified subcritical multiplication equation but indicators other than the high source range neutron flux alarm are credited to alert the operator of a possible inadvertent boron dilution. By crediting the other indicators the Modes 3-5 boron dilution times satisfy the 15 minute SRP 15.4.6 acceptance criteria. Therefore, Question 15.4.6-9 is acceptable and closed.

The boron concentration C2 is cycle-dependent. Therefore, the staff asked RAI No. 682-5367, Question 15.4.6-7, how the applicant ensures that the C2 values in DCD, Tables 15.4.6-1, bound all future core designs and if C2 is checked in the reload safety methodology.

To address Question 15.4.6-7 [Reference 9], the applicant states that the DCD analysis is intended to bound more than the first operating cycle; however, the Mitsubishi Reload Evaluation Report, MUAP-07026-P [Reference 10], describes how key accident analysis inputs will be confirmed and evaluated for each operating cycle. The applicant further explains that MUAP-07026-P identifies the maximum critical boron concentration for Modes 1 through 6, the difference in boron concentration from initial condition to critical, boron worth versus

concentration, and available SDM as safety analysis input parameters that must be evaluated for each operating cycle for the DCD inadvertent boron dilution analysis. Based on the added discussion provided by the applicant, it is shown that the safety analysis parameters in question will be re-analyzed for each new cycle despite providing what is intended to be a bounding analysis in DCD Section 15.4.6. This is also implied since the cycle-specific COLR, as part of the TS, includes the mode-specific SDM requirements, which include the initial and critical boron concentrations as part of this calculation. Consequently, Question 15.4.6-7 is closed.

With respect to core parameters such as reactor power, coolant average temperature, and minimum DNBR, the applicant explained the boron dilution event is bounded by an uncontrolled RCCA withdrawal and those results have been shown to be acceptable. In RAI 311-2347, Question 15.4.6-3, the staff asked MHI to provide detailed information on the rate of reactivity insertion for both the boron dilution event and the uncontrolled rod withdrawal at power in order to support the claim that the boron dilution event from power conditions is bounded by the uncontrolled rod withdrawal at power. In response to Question 15.4.6-3 [Reference 8], the applicant compares the uncontrolled rod withdrawal analysis reactivity insertion rates to the boron dilution reactivity insertion rate. The boron dilution reactivity insertion rate falls within the range analyzed, and therefore the analysis is considered to be bounded by the uncontrolled rod withdrawal at power. Question 15.4.6-3 is closed.

Open Items

OI 15.4-8. RAI 902-6318, Question 15.4.6-10 is a follow-on to RAI 682-5367, Question 15.4.6-6, regarding the ability to dilute the RCS under operational Modes 4 and 5, no RCPs running, and Mode 6, refueling.

15.4.6.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD that pertain to this section.

15.4.6.6 Conclusions

As a result of the open and confirmatory items, the staff is unable to finalize its conclusion on Section 15.0 in accordance with GDCs 10, 13, 15, and 26 requirements.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Introduction

This section documents the NRC staff evaluation of the applicant's DCD Tier 2 analysis of the US-APWR system response to a postulated inadvertent loading and operation of a fuel assembly in an improper position. The general design requirements discussed in Section 15.4.7 of NUREG-0800 were used to determine that the US-APWR design is in compliance with the Commission's regulations.

Analysis of the inadvertent loading and operation of a fuel assembly in an improper position requires consideration of a spectrum of misloading events. The applicant must identify the limiting misload event that is undetectable by in-core instrumentation. The kinds of errors that should be considered include loading of one or more fuel assemblies into improper locations and, where physically possible, with incorrect orientation. If burnable poison or fuel rods are added to or removed from fuel assemblies, errors in these processes must also be considered.

The applicant is also responsible for identifying changes in the power distribution in addition to increased local power density that may result from an inadvertent loading and operation of a fuel assembly in an improper position. There should also be provisions made to search for loading errors at the beginning of each fuel cycle.

Finally, the applicant should consider the effect of misloaded fuel on nuclear design parameters, the detection of fuel-loading errors, and any operational restrictions that would assist in staying with fuel rod failure limits.

15.4.7.2 Summary of Application

DCD Tier 1: There were no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant provided a DCD Tier 2 description in Section 15.4.7, summarized here as follows:

The barriers in place to mitigate the inadvertent loading and operation of a fuel assembly in an improper position are discussed citing multiple checks that take place before full power operation occurs including low power testing and/or power ascension testing. A loading error that leads to a larger increase in power peaking can be detected by the in-core instrumentation that provides core mapping and temperature measurement prior to full power operation. Consequently, loading errors that result in larger-than-expected power peaking can be discovered and corrected at this point. The availability and adequacy of instrumentation and controls is described in DCD Tier 2 Section 15.0.0.3. Mechanical constraints are also in place to prevent a situation where a fuel assembly is in the correct location but has an incorrect azimuthal alignment.

No transient occurs for this event, thus the typical transient analysis codes are not used. The ANC code is used to calculate both a normal expected radial power distribution and the radial power distributions resulting from the four possible types of fuel loading errors identified.

The results of the power distribution analysis for the fuel loading errors show differences between the measured and predicted power distributions that are abnormally large indicating an obvious error in fuel loading. Since these measurements are performed at low power, 30 percent in this case, the vast majority of fuel loading errors can be detected before the core reaches a high power level. For the other fuel loading errors that do not provide obvious indication that fuel has been improperly loaded in the core, core analyses conservatively include an 8 percent allowance for uncertainties in local power peaking.

This event involves only changes in the distribution of power and heat flux within the core. Overall core power, RCS flow, and RCS pressure are not changed. Therefore, the maximum reactor coolant pressure remains well below 110 percent of the design pressure and the integrity of the RCPB is maintained.

The radiological consequences of this event are bounded by the radiological consequences of the rod ejection accidents evaluated in DCD Tier 2 Section 15.4.8.5.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.4.7.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (hereafter referred to as NUREG-0800 or the SRP), Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.7.

1. General Design Criterion (GDC) 13, as it relates to providing instrumentation to monitor variables over anticipated ranges for normal operations, for anticipated operational occurrences, and for accident conditions.

2. 10 CFR Part 100, as it relates to offsite consequences resulting from reactor operations with an undetected misloaded fuel assembly.

The primary safeguards against fuel-loading errors are procedures and design features to minimize the likelihood of the event. Additional safeguards include incore instrumentation systems that would detect errors. However, should an error be made and go undetected, it is possible in some reactor designs for fuel rod failure limits to be exceeded. Therefore, the following acceptance criteria cover the event of operation with misloaded fuel caused by loading errors:

1. To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations.

2. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 criteria. A small fraction is interpreted to be less than 10 percent of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error should include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

15.4.7.4 Technical Evaluation

The applicant discusses various measures in place to detect the inadvertent loading and operation of a fuel assembly in an improper position. The scope of the inadvertent fuel

assembly loading accident considers the misload of either a fuel assembly or in-core control component (ICCC) in which the fuel assembly or ICCC is loaded into an incorrect position in the core. The applicant explains that due to the strictly managed administrative procedures for fuel assembly loading, as described in DCD Tier 2 Chapter 14, the event is highly unlikely to occur. Multiple operators match identification numbers corresponding to fuel assemblies and ICCCs with a core loading diagram as the various elements are placed within the core. After fuel loading is complete, the identification numbers are checked again to assure that all the fuel assemblies are loaded correctly. An in-core power distribution measurement, at low power, is then performed. Additionally, mechanical constraints are in place to prevent a situation where a fuel assembly is in the correct location, but is rotated in the incorrect azimuthal orientation.

In a letter dated April 2, 2009, the staff issued RAI 312-2348, Question 15.4.7-1, asking the applicant to provide clarification regarding the mechanical constraints and the possibility of human error that could lead to inadvertent azimuthal rotation. In a letter dated May 15, 2009, the applicant responded to Question 15.4.7-1 [Reference 11] by stating that the purpose of the mechanical constraints is to allow the fuel to be loaded in only one possible azimuthal orientation. The applicant references DCD Tier 2 Figure 4.2-6, "Schematic View of Top Nozzle" showing two alignment holes in opposing corners and an indexing hole in a third corner. Based on the figure presented, it is clear to the staff that only a single rotational orientation is possible, making a rotational misload impossible. Question 15.4.7-1 is consequently closed.

Four fuel assembly misload scenarios, described in DCD Tier 2 Section 15.4.7.3.2, "Input Parameters and Initial Conditions" are identified by the applicant, which include:

1. An assembly interchange with a large reactivity difference

2. An assembly interchange with a small reactivity difference

3. An assembly interchange with and without burnable absorber

4. A burnable absorber loaded in an incorrect location

A demonstrative example used the NRC approved ANC code to compare low-power maps for a correctly loaded core (predicted) to low-power maps for an incorrectly loaded core (measured). In RAI No. 312-2348, Question 15.4.7-2, the staff asked the applicant to provide a reference in DCD Tier 2 Section 15.4.7 for the code version of ANC used in the analysis for the section. The applicant responded [Ref 11] by stating that the approved ANC methodology is described in detail in DCD Tier 2 Section 4.3.3.1, "Nuclear Design Methods," and no changes have been made to the approved methodology described in the corresponding topical reports listed as references 4.3-12, 4.3-14, and 4.3-15 in DCD Tier 2 Section 4.3.6, "References." Since the code is capable of calculating power distribution maps, and was previously approved by the NRC, the staff finds its use for the calculations performed in DCD Tier 2 Section 15.4.7.3 acceptable, and Question 15.4.7-2 is closed.

The results, shown in DCD Tier 2 Figures 15.4.7-1, "Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core, Case A: Assembly Interchange with a Large Reactivity Difference"; 15.4.7-2, "Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core, Case B: Assembly Interchange with a Small Reactivity Difference"; 15.4.7-3, "Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core, Case C: Assembly

Interchange with and without Burnable Absorber"; and 15.4.7-4, "Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core, Case D: Burnable Absorber Loaded in Incorrect Location," show large deviations (maximum deviations of 49.2 percent, 9.2 percent, 40.8 percent, and 38.9 percent, respectively) between predicted and measured power maps, which clearly indicate a potential fuel assembly misload for the selected scenarios. In DCD Tier 2 Section 15.4.7.2, "Sequence of Events and Systems Operation" the applicant mentions that core analyses include an 8 percent allowance for uncertainties in local power peaking. This margin is important to cover potential scenarios not considered by the applicant, which could lead to non-obvious or imperceptible fuel assembly misload events. In RAI 312-2348, Question 15.4.7-3, the staff asked the applicant to elaborate on the sensitivity of the in-core instrumentation used throughout startup to detect deviations from the predicted power distribution. The staff also asked about the magnitude of the minimum detectable deviation. The applicant responded by referencing Section 3.0 of MUAP-07021-P, "US-APWR Incore Power Distribution Evaluation Methodology," [Ref 12] which describes how the in-core instrumentation system measures relative in-core power distributions; however, the sensitivity of the in-core instrumentation was not explicitly addressed in the response. It was also stated that the maximum deviation is compared to a 10% assessment criteria and, if a larger deviation occurs relative to this criteria, the plant would be placed in a safe condition to evaluate the discrepancy before power ascension. Smaller deviations would be covered by the 8 percent power distribution uncertainty. The applicant provided a measured versus predicted power distribution difference criterion [Reference 12] which would cause an evaluation of a potential fuel assembly loading error. The maximum positive difference (i.e., measured power greater than predicted), is a safety concern as DNBR margin would be less than expected. Therefore, the applicant has provided the necessary information to address the safety concern and Question 15.4.7-3 is closed.

It is clear to the staff that there are multiple barriers in place to prevent the inadvertent loading and operation of a fuel assembly in an improper position including: (1) strictly managed administrative procedures, (2) verification using identification numbers by multiple operators as fuel is being loaded, (3) final verification using identification numbers after fuel is loaded, (4) in-core power distribution measurement at low power, and (5) mechanical constraints. The staff agrees that these barriers are sufficient to meet the requirements of GDC 13, which requires adequate provisions to minimize the potential of a misloaded fuel assembly going undetected.

15.4.7.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD that pertain to this section.

15.4.7.6 Conclusions

The staff has evaluated the consequences of a spectrum of postulated fuel loading errors. The staff concludes that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable), or the error is undetectable but the offsite consequences of any fuel rod failures are a small fraction of 10 CFR Part 100 criteria. The radiological consequences of this event are bounded by the rod ejection accidents evaluated in DCD Tier 2 Section 15.4.8.5, "Radiological Consequences" as the applicant has committed to low power flux maps and evaluating a 10 percent measured versus predicted assembly power difference in a safe plant condition. These two commitments, combined with an 8 percent peaking factor uncertainty,

would ensure that an undetected misloading would result in the number of failed fuel rods less than the assumed 10 percent in the rod ejection analysis. Therefore, the staff concludes that the requirements of GDC 13 and 10 CFR Part 100 have been met.

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Introduction

The review of rod ejection accidents considers the consequences of a control rod ejection accident and whether the fuel damage from such an accident could impair cooling water flow. The review covers the applicant's description of the occurrences that lead to the accident, initial conditions, rod patterns and worth, safety features designed to limit the amount of reactivity available, the rate at which reactivity can be added to the core, and methods for analyzing the accident. The review also examines potential fission product releases from a rod ejection accident. The radiological consequences are discussed in Section 15.4.8.5 of this SE.

15.4.8.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.4.8, summarized here in part, as follows:

The application in this section addresses an accident in which there is a mechanical failure of a control rod drive mechanism (CRDM) housing. This failure results in the ejection of a rod cluster control assembly (RCCA) and its drive shaft. The consequence of this RCCA ejection includes rapid positive reactivity insertion with an increase of core power peaking, possibly leading to localized fuel rod failure. This nuclear power excursion is terminated by Doppler reactivity feedback from increased fuel temperature, and the core is shut down by the high power range neutron flux trip, over temperature ΔT, or low pressurizer pressure trip.

For large reactivity insertions, the event is terminated by Doppler reactivity feedback followed by a high neutron flux trip. For low ejected rod worths, the core average power may not reach the high neutron flux trip setpoint. If this occurs core power will initially increase to greater than 100 percent and the RCS pressure will decrease as a function of the ejected rod hole size. Increasing core power with a decreasing RCS pressure could lead to a percentage of DNB fuel failures greater than the large reactivity insertion ejections. As such, the applicant evaluated various low ejected rod worth scenarios.

US-APWR DCD Table 15.4.8-1, “Time Sequence of Events for Rod Ejection” addresses four cases as follows:

- Case 1 HFP, BOC
- Case 2 HFP, EOC
- Case 3 HZP, BOC
- Case 4 HZP, EOC

The following automatic trip signals are assumed in the rod ejection analyses:

- High power range neutron flux (high setpoint)
- High power range neutron flux (low setpoint)
- Over temperature ΔT trip
- Low pressurizer pressure trip

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER

15.4.8.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, the associated acceptance criteria, and the review procedures are given in Section 15.4.8 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.4.8 of NUREG-0800.

The relevant requirements and corresponding acceptance criteria are:

1. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

2. GDC 28 requires reactivity control systems designed with appropriate limits on potential reactivity increases so the effects of a rod ejection accident can result in neither damage to the RCPB nor sufficient disturbance to impair the core cooling capability.

   GDC 28 requirements apply to this section because the reviewer evaluates the maximum reactor pressure during any portion of the transient corresponding to a rod ejection. ASME Codes provide guidance for the acceptability of anticipated accident pressure. The review also examines the extent of fuel damage from a rod ejection accident. RG 1.77 and Section 4.2 of NUREG-0800 provide guidance for acceptability of anticipated core damage.

   This criterion provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired by a control rod ejection accident.

3. 10 CFR 100.11 or 10 CFR 50.67 requires that the exclusion area and the low population zone be defined by assurances that specified limits for postulated fission product releases will not be exceeded in radiation doses to individuals at the outer boundaries of those regions.

These requirements apply to this section because rod ejections are included among the potential accidents for which fission product releases are postulated. Review under NUREG-0800 Section 15.0.3 determines the source term used by the reviewer.

These requirements provide assurance that offsite radiation doses from a pressurized water reactor rod ejection accident will not exceed guideline doses specified in 10 CFR 100.11 or 10 CFR 50.67.

Review of the applicant's analyses to meet the above requirements should consider:

1. For Requirements 1 and 2 above:

   A. A spectrum of initial conditions, which must include zero, intermediate, and full-power, is considered at the beginning and end of a reactor fuel cycle for examination of upper bounds on possible fuel damage. At-power conditions should include the uncertainties in the calorimetric measurement.

   B. From the initial conditions, considering all possible control rod patterns allowed by technical specification/core operating limit report power-dependent insertion limits, the limiting rod worths are determined. Where confirmation is necessary, the reviewer may calculate, as an audit, the worth of limiting rods.

   C. Reactivity coefficient values of the limiting initial conditions must be used at the beginning of the transient. The reviewer checks the reactivity coefficient curves used by the applicant with those reviewed under NUREG-0800 Section 4.3. The Doppler and moderator coefficients are the two of most interest. If there is no three-dimensional space-time calculation, the reactivity feedback must be weighted conservatively to account for the variation in the missing dimension(s).

   D. The reviewer inspects the control rod insertion assumptions, which include trip parameters, trip delay time, rod velocity curve, and differential rod worth. Trip parameters and delay time are reviewed under NUREG-0800 Section 7.2. Control rod worth is checked by the reviewer for consistency with the review under NUREG-0800 Section 4.3.

   E. The applicant's analytical methods are reviewed. The reviewer may use the results of previous case work if the analytical methods have been reviewed and approved by the staff. Otherwise, he/she must do a de novo review. Alternatively, the reviewer may audit several calculations, using methods acceptable to the staff (or staff consultants). The reviewer's primary concern is how well the elements of the analytical model represent the true three-dimensional problem. The reviewer also checks feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

2. For Requirement 3, the number of fuel rods with clad failure is determined (for use by NUREG-0800 Section 15.0.3 reviewer in evaluating radiological consequences of the rod ejection accident) by the following procedure:

   A. The reviewer determines whether an acceptable procedure for calculating a departure from nucleate boiling (DNB) condition during the reactivity excursion is

used. This determination may be done by reference to previous cases for the same nuclear steam supply system vendor. If no approved technique is available (e.g., the first project using a new or substantially revised model), the reviewer must perform a separate detailed review, which may be documented separately in a topical report. DNB must be calculated in accordance with the criteria reviewed and accepted under NUREG-0800 Section 4.4. Typically, the criteria define a DNB ratio (DNBR) less than 1.30 when NRC-approved critical heat flux correlations are used.

B. The reviewer must determine the total number of failed rods used in the radiological evaluation. The number of fuel rod failures due to each failure mechanism addressed in NUREG-0800 Section 4.2 must be combined.

C. The reviewer determines the acceptability of the time-dependent activity releases from both containment leakage and plant cool-down (steaming/release via atmospheric dump valves). Each scenario should be investigated in combination and separately for the most severe release path.

15.4.8.4 Technical Evaluation

Rod ejection accidents (REAs) are initiated by the mechanical failure of a control rod drive mechanism (CRDM) housing and are classified as PAs because the mechanical design reduces the probability of this occurring. The accident is of interest from HZP to HFP conditions. The ejection of the control rod drive and RCCA leads to a reactivity insertion and a power excursion. Doppler reactivity feedback limits the extent of energy deposition from the power excursion and a reactor trip on high power range neutron flux trip provides complete shutdown. The high power range neutron flux rate trip is conservatively ignored. For low ejected rod worths, which don't reach the high neutron flux trip, the over temperature ΔT or low pressurizer pressure trips the reactor. The limiting single failure for this event is the failure of one train of the RTS. Any one of the remaining trains is adequate to provide the protection functions credited in the analysis.

The event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could cause a LOOP. However, the resulting RCP coastdown would not start until after the time of peak radial average fuel enthalpy, peak fuel temperature, peak reactor coolant pressure, and minimum DNBR values, for the entire transient, are the same whether offsite power is available or unavailable.

The analysis of these events is carried out for HFP and HZP conditions at both BOC and EOC. The following assumptions, many of which are used to obtain conservative results, are used for the large reactivity insertions caused by high ejected rod worths:

1. Initial conditions assume a 24-month equilibrium core. HFP assumes 102 percent of rated power and both HFP and HZP assume initial reactor coolant temperature 2.2ºC (4ºF) above the nominal value and pressurizer pressure 0.21 MPa (30 psi) below the nominal value.

2. Control rods are assumed to be initially at their insertion limits.

3. A conservative ejected rod reactivity worth at the design limit is inserted in 0.1 s. This is imposed on the analysis when TWINKLE-M is used in either a one-dimensional (1-D) or three-dimensional (3-D) mode. When TWINKLE-M is used in a 3-D mode, the position of the highest worth RCCA is used; in the 1-D model this does not enter into consideration.

4. Doppler reactivity feedback is conservatively estimated by multiplying the fast absorption cross section for the given change in the calculated fuel effective temperature by a conservative multiplier. However, in the one-dimensional methodology, a small Doppler weighting factor (>1.0) is used to compensate for collapsing the 3-D problem into a 1-D axial model.

5. Moderator reactivity feedback is conservatively estimated. It has a small effect after the power peaks when sufficient time has elapsed for heat to be transferred to the coolant.

6. The hot spot fuel calculation in VIPRE-01M conservatively assumes DNB; heat transfer is therefore calculated using the Bishop-Sandberg-Tong correlation for film boiling heat transfer after DNB.

7. Conservative assumptions for trip reactivity, rod drop time, and RTS signal processing delay are used. Reactor trip reactivity used is the design limit, which is -4% Δk/k for the HFP case and -2% Δk/k for the HZP case.

8. Minimum delayed neutron fraction and minimum neutron lifetime are used.

9. The pellet and cladding gap conductance in the transient analysis with VIPRE-01M remains constant for fuel temperature and enthalpy analysis; instantaneously decreases to zero for an adiabatic fuel enthalpy analysis; rapidly increases to the maximum value for the cladding temperature analysis; and realistically increases for RCS pressure analysis.

10. HZP cases are assumed to trip on high power range neutron flux, low setpoint signal.

11. HFP cases are assumed to trip when the measured neutron flux reaches the high range neutron flux high sepoint plus uncertainity, including a single failure of one ex-core detector.

For the HFP cases, a one-dimensional TWINKLE-M model was used to determine power vs. time for use in VIPRE-01M. For BOC the bounding ejected rod worth used is 110 pcm and at EOC it is 120 pcm. The bounding design hot channel factors used are 5.0 and 6.0 at BOC and EOC, respectively. The applicant uses bounding values for analysis conservatism and margin for future core designs. The hot channel factor used in VIPRE-01M is assumed to instantaneously increase to the bounding value and is conservatively assumed to remain constant, ignoring feedback effects during the transient.

For the HZP cases, a three-dimensional TWINKLE-M model was used to determine power vs. time and hot channel factor for use in VIPRE-01M. For conservatism the maximum value of the hot channel factor used in VIPRE-01M is adjusted to the design limit. For BOC the bounding ejected rod worth used is 600 pcm and at EOC it is 800 pcm. The bounding design hot channel factors used are 14.0 and 35.0 at BOC and EOC, respectively. The applicant uses bounding values for analysis conservatism and margin for future core designs.

RAI 313-2361, Question 15.4.8-2:

> Values of ejected rod worth and hot channel factors used in the REA analysis are stated to be conservative. What are realistic values for these quantities for the events from zero and full power for both beginning- and end-of-cycle?

In response to Question 15.4.8-2 the applicant provided [Reference 13] a comparison table of BOC and EOC, HFP and HZP best estimate plus uncertainty ejected rod worths and hot channel factors with those assumed in the DCD. [(Proprietary information withheld under 10 CFR 2.390)]. Thus, the applicant demonstrated that the DCD analysis is conservative and therefore acceptable.

The staff noted that the DCD only evaluated BOC and EOC at HFP and HZP conditions. According to SRP 15.4.8 intermediate powers should also be evaluated.

RAI 313-2361, Question 15.4.8-5:

> Per Regulatory Guide 1.77, perform rod ejection analyses for both beginning of cycle and end of cycle starting from a low-power condition and provide analysis results.

In response to Question 15.4.8-5 the applicant provided [Reference 13] 3-D, TWINKLE-M, best estimate calculations at 0, 20, 40, 60, 80 and 100 percent power. The applicant compared the best estimate fuel centerline temperature and fuel enthalpy rise calculations with those in the DCD. [(Proprietary information withheld under 10 CFR 2.390)]. The margin between the DCD and best estimate calculation is primarily due to the differences in ejected rod worths and hot channel factors. The applicant demonstrated that by using the conservative DCD assumptions only HFP and HZP cases need to be evaluated. The response to Question 15.4.8-5 is acceptable and this RAI is closed.

In the DCD the applicant does not provide HFP, prompt fuel enthalpy rise results. Therefore, the staff asked RAI 313-2361, Question 15.4.8-7.

RAI 313-2361, Question 15.4.8-7:

> In accordance with SRP Section 15.4.8 guidance found in Part III, “Review Procedures,” include consideration of PCMI failure during the rod ejection analysis for at-power conditions.

The applicant noted [Reference 13] that HFP fuel enthalpy rise is very low based on the calculations performed in response to RAI 313-2361, Question 15.4.8-5. The calculated value is significantly below the minimum acceptable value of 60 cal/g given in SRP Section 4.2, “Fuel System Design,” Appendix B. It is expected that the lower power cases bound the HFP cases as the ejected rod worths and assumed peaking factors are greater for powers less than HFP. The applicant demonstrated that the HFP fuel enthalpy rise is not limiting and significant margin to the minimum limit exists. The staff finds the response to Question 15.4.8-7 acceptable and Question 15.4.8-7 is closed.

As part of the REA method the applicant stated that a realistic VIPRE-01M gap conductance model is used to calculate DNB and the peak RCS pressure response. The staff asked the following question regarding the DNB and RCS pressure gap conductance model.

RAI 313-2361, Question 15.4.8-8:

> Provide the specific "realistic" gap conductance models employed for the DNB and RCS pressure analysis along with the justification for their applicability.
>
> Provide the basis that the Ross-Stoute gap conductance model is acceptable for use in the high ejected rod worth and peak RCS pressure cases**; Open Item 15.4-9.**

In RAI 785-5885, Question 15.4.8-11 the staff asked for the basis of the uncertainty used in the power range high neutron flux setpoint and the impact on the number of rods in DNB if a low worth control rod that was ejected which did not reach the high flux setpoint.

In the HFP rod ejection accident analysis, the applicant applied a 9 percent uncertainty to the power range high neutron flux (high setpoint) reactor trip setpoint. This means that within the simulation, the reactor tripped at a calculated power of 118 percent, rather than the nominal trip setpoint of 109 percent. However, as described in MUAP-09022 [Reference 14], this 9 percent uncertainty was derived for AOO conditions in which the core power distribution was varying relatively slowly, and may not be appropriate for rapid transients. Therefore, the staff issued RAI 785-5885, Question 15.04.08-11, requesting that the applicant justify this choice of uncertainty.

RAI 785-5885, Question 15.4.8-11:

> In the DCD rod ejection analysis, the analytical limit for power range neutron flux (high setpoint) is 118 percent, which includes the nominal setpoint of 109 percent plus 9 percent additional uncertainty. As described in MUAP-09022 and RAIs associated with MUAP-07010-P, this 9 percent bounds the uncertainty in power distribution effects for AOOs, but may not bound the uncertainty for rapid reactivity insertions such as control rod ejection. Justify 9 percent uncertainty as being appropriate for the rod ejection analysis, or determine what the appropriate uncertainty should be and revise the rod ejection analysis accordingly.

In response [Reference 15], the applicant performed sensitivity calculations at BOC and EOC in which the base case (reactor power tripped at 118 percent calculated power) was compared against a case in which the reactor tripped when the third-highest “measured” ex-core detector signal reached a power level that conservatively included some uncertainty, but neglected uncertainty due to power distribution effects. The “measured” ex-core detector signals were calculated using weighting factors derived from neutron transport calculations with the well-established DORT code. The requirement that the third-highest ex-core detector signal reaches the trip setpoint accounts for a possible failure of one other ex-core detector. The results indicate that for both BOC and EOC conditions there would be a very small delay in the reactor trip relative to the base case (less than 0.01 seconds), and therefore the figures of merit (peak reactor power, local enthalpy deposition, and peak fuel centerline and cladding temperatures) would not be significantly impacted. Nevertheless, the applicant has revised the analysis methodology to explicitly account for the response of the ex-core detectors.

The staff has reviewed the applicant's response, and finds that explicit consideration of the ex-core detectors has been appropriately addressed. The staff also finds the use of a more detailed, physics-based consideration of the response of the reactor protection system to be appropriate for this accident scenario. Therefore, the staff considers RAI 785-5885, Question 15.4.8-11 closed.

For low ejected rod worths that are not terminated by reactor trip or mitigated by Doppler feedback, the primary concern is the number of fuel rods that may experience DNB. The applicant analyzed low-ejected rod worths by evaluating the accident with three, 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. Details of the methodology and codes used are described in the "Non-LOCA Methodology Topical Report" [Reference 1].

The applicant addressed DNB evaluation method and results for low ejected rod worth cases in Question REA-12 of the Non-LOCA Topical Report [Reference 1]. The applicant addressed the need to document the results in the DCD in response to RAI 785-5885, Question 15.4.8-11 [Reference 15]. The limiting event was the long-term, slow depressurization scenario but the number of rods in DNB remained below the 10 percent assumed in the radiological consequence analysis.

The calculation of RCS pressure during these events uses the results from TWINKLE-M for reactor power in VIPRE-01M. The latter then generates core total void fraction and heat flux for use in MARVEL-M vs. time. The following assumptions are used:

1. The initial power level for HFP is conservatively assumed to be 102 percent of rated power and for all cases the initial reactor coolant temperature is 2.2ºC (4ºF) above the nominal value and pressurizer pressure 0.21 MPa (30 psi) below the nominal value.

2. No pressurizer spray is assumed.

3. The void fraction calculated by VIPRE-01M is conservatively multiplied by a factor for use in MARVEL-M.

The gap conductance model can affect the peak RCS pressure as described in the response to RAI 313-2361, Question 15.4.8-8. The staff has been unable to determine if the current response conservatively predicts conservative peak RCS pressure and an **Open Item 15.4-9** has been created (see above).

The calculated peak radial average fuel enthalpy was less than 150 cal/g thereby satisfying the peak radial average fuel enthalpy criteria of SRP 4.2, Appendix B. Likewise, the maximum enthalpy rise was below 60 cal/g hence satisfying SRP 4.2, Appendix B, "Figure B-1: PWR PCMI Fuel Cladding Failure Criteria." The calculated fuel centerline temperatures are 2333ºC (4232ºF) and 2395ºC (4343ºF) for BOC and EOC, respectively, which are below the minimum melting temperature including the effects of fuel burnup. The number of rods calculated to fail due to DNB was less than the 10 percent assumed in the radiological consequence analyses.

The peak RCS pressure analysis demonstrated that the RCS remains below the acceptance criterion of 110 percent of the system design pressure.

Open Items

OI 15.4-9. RAI 911-6326, Question 15.4.8-12. Provide the basis that the Ross-Stoute gap conductance model is acceptable for use in the high ejected rod worth and peak RCS pressure cases.

15.4.8.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD that pertain to this section.

15.4.8.6 Conclusions

As a result of the open item, the staff is unable to finalize its conclusion on Section 15.4.8 in accordance with the requirements of GDC 13 and 28.

15.4.8.7 References

1. MUAP-07010, “Non-LOCA Methodology,” Rev. 4, May 2012, ML12146A090
2. Non-LOCA Methodology SE – this is an OI
3. MUAP-07009, “Thermal Design Methodology.” Rev 0, May 2007, ML071520271
4. MUAP-07009, Thermal Design Methodology, SE – this is an OI
5. UAP-HF-09344, “MHI's Response to US-APWR DCD RAI No. 308-2340 Revision 2,” July 2009, ML091890023.
6. UAP-HF-09229, “MHI's Response to US-APWR DCD RAI No. 309-2345 Revision 1,” May 2009, ML091390655.
7. UAP-HF-09345, “MHI's Response to US-APWR DCD RAI No. 310-2346 Revision,” July 2009, ML091890111.
8. UAP-HF-09303, “MHI's Response to US-APWR DCD RAI No. 311-2347 Revision 1,” June 16, 2009, ML091690076
9. UAP-HF-11104, “MHI's Responses to US-APWR DCD RAI No. 682-5367 Revision 0 and RAI No. 708-5455 Revision 2,” April, 2011, ML11108A065
10. MUAP-07026, “Mitsubishi Reload Evaluation Methodology”, Rev. 1, August 2011, ML11266A126
11. UAP-HF-09230, “MHI's Response to US-APWR DCD RAI No. 312-2348 Revision 1,” May, 2009, ML091390654
12. MUAP-07021-P, “US-APWR Incore Power Distribution Evaluation Methodology,” Rev. 0, December 2007, ML08025009
13. UAP-HF-09346, “MHI's Response to US-APWR DCD RAI No. 313-2361” Revision 2, July 2009, ML091890113
14. MUAP-09022, “US-APWR Instrument Setpoint Methodology,” Rev. 2, May 2011 ML11160A138
15. UAP-HF-11276, “MHI's Response to US-APWR DCD RAI No. 785-5885,” Revision 3, August 2011, ML112450423

15.5 Increase in Reactor Coolant Inventory

15.5.1 Inadvertent Operation of ECCS and CVCS Malfunction that Increase Reactor Coolant Inventory

15.5.1.1 Introduction

This section documents the staff's review of DCD Tier 2, Sections 15.5.1 "Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory," and 15.5.2 "Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory." Both of these events could cause an unplanned increase in reactor coolant inventory, which can fill the pressurizer with liquid. These events will be discussed together because both are AOOs that abide by the same requirements and acceptance criteria.

15.5.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Sections 15.5.1 and 15.5.2, summarized here as follows:

The application states that inadvertent operation of ECCS is not applicable to the US-APWR because no components of the ECCS are capable of injecting water into the RCS at normal operating pressures.

The CVCS malfunction is modeled as the full-open failure of the charging flow control valve, causing a net increase in coolant mass to the RCS, resulting in an increase in pressurizer level. While this event will be terminated by the automatic CVCS isolation function on high pressurizer level, the applicant included an evaluation that assumes this function is unavailable. The purpose of this analysis is to determine the time available after the high pressurizer water level alarm for the operator to perform actions to end the transient before the pressurizer fills.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.5.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 15.5.1-15.5.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.5.1-15.5.2 of NUREG-0800.

1. GDC 10, which requires that the reactor core and associated coolant control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

2 GDC 13, which requires, in part, that the effect of instrumentation shall be provided to monitor variables and systems over their anticipated ranges for AOOs to assure adequate safety. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, which requires that the RCS and its associated auxiliary control and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operations, including AOOs.

4. GDC 26, which requires, in part, the reliable control of reactivity changes to assure that SAFDLs are not exceeded under conditions of normal operation, including AOOs, with appropriate margin for malfunctions, such as stuck rods.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the ASME Boiler and Pressure Vessel Code.

2. Fuel cladding integrity should be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs.

3. An AOO should not generate a more serious plant condition without other faults occurring independently.

15.5.1.4 Technical Evaluation

The staff agrees that inadvertent operation of ECCS is not applicable because neither the safety injection pumps nor the accumulators have sufficient head to inject water into the RCS when it is at normal operating pressure. Hence, the only event that inadvertently increases reactor coolant inventory in the US-APWR is a CVCS malfunction.

The CVCS malfunction event credits the automatic CVCS isolation function on high pressurizer level to terminate the event (upon incorporation of Confirmatory Item 15.00-1). However, the analysis included in the DCD does not credit this action; instead it runs the transient out until the pressurizer fills in order to determine how much time would be available for the operator to manually end the transient. The staff will base its safety finding on the first scenario, noting that the applicant does not credit operator actions to mitigate this event.

The CVCS malfunction event was analyzed using the MARVEL-M computer code and methods described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER.

The CVCS malfunction presented in the DCD is the full-open failure of the charging flow control valve coupled with isolation of letdown flow because this scenario was found to cause the fastest increase in RCS liquid volume. The applicant stated that because this event is not

limiting with respect to fuel damage limits, the DCD only includes a case to evaluate peak pressurizer water volume.

In order to confirm this assertion, the staff requested a plot of DNBR in RAI 307-2336, Question 15.5.2-3. The applicant responded on June 16, 2009 with a curve demonstrating the DNBR limit is not challenged by this event.

The initial conditions (reactor power, RCS temperature and RCS pressure) are based on nominal values with uncertainties added in the direction to maximize pressurizer water volume as justified in the applicant's response to RAI 297-2287, Question 15.0.0-8. The pressurizer water level is initiated at the nominal level plus uncertainty rather than the maximum level allowed by TS 3.4.9. This is acceptable because the effectiveness of CVCS isolation to mitigate the event is not dependent on the initial pressurizer water level, nor is the time between the high level alarm and pressurizer fill. The staff confirmed that the remaining parameters used in the analytical model were suitably conservative, and that non-safety systems were only assumed operational if they adversely impact the results.

The analysis assumes LOOP is coincident with the reactor trip. The staff agrees this is conservative because the June 16, 2009, response to RAI 307-2336, Question 15.5.2-1, demonstrated that the pressurizer fills 30 seconds sooner with LOOP compared to no-LOOP. Additionally, the staff notes that this sequence of events conservatively ignores the 3-second delay between reactor/turbine trip and LOOP.

The analysis predicts a high pressurizer water level signal at 1062 seconds and, if the CVCS injection continues, the pressurizer fills 84 seconds later. The applicant stated that the CVCS injection will not continue because the US-APWR is designed to isolate CVCS on a high pressurizer water level signal. The staff will determine if the CVCS isolation valves close prior to pressurizer fill when the applicant provides the associated signal delays and valve closure times (Open Item 15.00-1, discussed in Section 15.0.0.4 of this SER).

The single failure assumed in this transient was one train of the RTS. The staff will determine if this is appropriate after the applicant provides a single failure assessment of CVCS isolation (Open Item 15.00-2, discussed in Section 15.0.0.4 of this SER).

Numerical results for the minimum DNBR, primary system pressure and secondary system pressure are included in the applicant's response to RAI 297-2287, Question 15.0.0-16, demonstrating that no acceptance limits are exceeded. No fuel failures are predicted; therefore, the radiological consequences for these events are bounded by the radiological consequences for the Section 15.1.5 MSLB, discussed in Section 15.0.3.4 of this SER.

15.5.1.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.5.1.6 Conclusions

As a result of the open items, the staff is unable to finalize its conclusion on Section 15.5 in accordance with the requirements of GDC 10, 15, and 26.

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a Pressurizer Pressure Relief Valve

15.6.1.1 Introduction

DCD Tier 2, Section 15.6.1 describes the analysis of the inadvertent opening of a pressurizer pressure relief valve, which could be caused by a spurious electrical signal or by an operator error. The event leads to a decrease of reactor coolant inventory and depressurization of the RCS. This event can occur one or more times during the plant's lifetime and is, therefore, classified as an AOO as defined in 10 CFR 50, Appendix A. This section describes the staff's evaluation of the event.

15.6.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.6.1, summarized here as follows:

The design basis event is assumed to be an inadvertent opening of a pressurizer depressurization valve (DV). This results in a decrease in RCS inventory and pressure until the event is mitigated by a reactor trip on a low pressurizer pressure signal.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.6.1.3 Regulatory Basis

The staff review of the event is based on the guidance specified in Section 15.6.1 of the SRP of NUREG-0800, which specifies the acceptance criteria of compliance with the following relevant requirements of the Commission regulations.

- GDC 10, as it relates to designing the RCS with appropriate margin to assure that SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, as it relates to providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation to assure adequate safety, and to providing appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, as it relates to designing the RCS and associated auxiliary systems with sufficient margin to assure that the design conditions of the RCPB are not exceeded during normal operations, including AOOs.

- GDC 26, as it relates to providing a reactivity control system capable of reliably controlling reactivity changes during manual operations and AOOs so that the SAFDLs are not exceeded.

- 10 CFR 52.47(a) and 52.79(a), as they relate to demonstrating compliance with any technically relevant portions of the Three Mile Island (TMI)-related requirements set forth in 10 CFR 50.34(f)(1)(vi) and 10 CFR 50.34(f)(1)(iii), for DC and COL reviews.

SRP 15.6.1 specifies the following acceptance criteria adequate to meet the above requirements:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design.

- Fuel cladding integrity should be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit.

- An AOO should not develop into a more serious plant condition without other faults occurring independently.

15.6.1.4 Technical Evaluation

An accidental depressurization of the RCS could occur by the inadvertent opening of a pressurizer pressure relief valve or similar valve. The US-APWR pressurizer design includes spring-loaded safety relief valves (SRVs), motor-operated safety depressurization valves (SDVs), and a motor-operated depressurization valve (DV) used for the mitigation of severe accidents. Since a DV has more relief capacity than an SRV or an SDV, the design basis event is assumed to be an inadvertent opening of a DV because it will result in a more rapid decrease of reactor inventory and depressurization, as well as the most severe core conditions.

The applicant performed the analysis using the MARVEL-M computer code and methods described in MUAP-07010-P. NRC approval of MUAP-07010-P is described in Section 15.0.2.4 of this SER. The analysis also applies the RTDP statistical method for the DNBR analysis, as described in DCD Section 4.4.2.2.1. Consistent with the use of RTDP, the initial values of the reactor power, coolant temperature, and the RCS pressure are assumed to be the nominal values. The primary coolant blowdown rate is assumed to be 120 percent of the rated capacity of one DV. The limiting single failure was determined to be the failure of one train of the reactor trip system. The analysis conservatively assumes the bounding minimum moderator density coefficient and maximum Doppler power coefficient. The reactor trip was initiated by the low pressurizer pressure. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive RCCA held out of the core. During the transient, the reactor RCS pressure rapidly decreases, which causes a decrease in power because of the moderator density reactivity feedback. The rod control system is assumed to be in the automatic mode to maintain the core at full power until a reactor trip. This assumption results in a more severe transient than if the rod control system was not in automatic mode. The staff finds these assumptions suitably conservative and acceptable.

The results of the analyses are shown in Figures 15.6.1-1 “Reactor Power versus Time, Inadvertent Opening of a Depressurization Valve” through 15.5.6-7 “DNBR versus Time,

Inadvertent Opening of a Depressurization Valve" of the DCD. The results show that the inadvertent opening of a DV results in a decrease in reactor coolant inventory and RCS pressure. The rod control system responds by maintaining power and average coolant temperature until the reactor trips. The low pressurizer pressure limit is reached at 28.3 seconds, with the reactor trip initiated 1.8 seconds later.

The reactor power remains at full power until the reactor trip occurs. The DNBR decreases initially, but increases rapidly following the reactor trip. The minimum DNBR occurs at 30.5 seconds and is well above the 95/95 safety limit. The DNBR remains above the 95/95 limit throughout the transient; therefore, fuel integrity is not degraded. The analysis is performed with the offsite power available. Since a LOOP and subsequent reactor coolant pump coastdown would not occur until after the reactor trip, the rapid decrease in the heat flux after the reactor trip compensates for the decrease in the RCS flow caused by the LOOP, and the minimum DNBR occurs before the initiation of a LOOP. Therefore, a LOOP has no effect on the calculated minimum DNBR.

This is a depressurization event as shown in Figure 15.6.1-3 "RCS Pressure versus Time, Inadvertent Opening of a Depressurization Valve," which shows that the RCS pressure decreases from the initial value. Since a breach in the RCPB is the initiating condition for this event, the maximum RCS pressure for this transient is the initial RCS pressure, which is assumed to be the maximum nominal RCS pressure. The staff concludes that the calculated results are in satisfaction of the specified acceptance criteria and are acceptable.

15.6.1.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2, that affect this section.

15.6.1.6 Conclusions

The applicant evaluated this transient using the method reviewed and found acceptable by the staff as discussed in 15.0.2 of this SER. The input parameters for this model were reviewed and found suitably conservative. The results showed SAFDLs maintained by minimum DNBR not below the 95/95 limit and a maximum pressure within the reactor coolant and main steam systems not in excess of 110 percent of the design pressures. The applicant has shown that this AOO would not develop into a PA without other faults occurring independently. Therefore, the staff concludes that the relevant requirements of GDCs 10, 13, 15, and 26 are met.

15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

This section of the DCD is evaluated above in Section 15.0.3.4.7 of this SER.

15.6.3 Radiological Consequences of Steam Generator Tube Failure

15.6.3.1 Introduction

This section documents the staff's review of DCD Tier 2, Sections 15.6.3 "Radiological Consequences of Steam Generator Tube Failure." An SG tube rupture (SGTR) event is considered a PA. The principal acceptance criterion for this event is to maintain the radiological releases below acceptable limits provided in 10 CFR Part 100. A secondary criterion is to prevent overfill of the SG secondary in order to prevent water from entering the steamlines.

15.6.3.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.6.3, summarized here as follows:

In the SGTR event, complete severance of a single SG tube is assumed to occur at full power. This event leads to leakage of radioactive coolant from the RCS to the secondary system, from which a release to the environment can occur. The operator is expected to recognize the occurrence of a SGTR event, to identify and isolate the ruptured SG and to take appropriate actions to stabilize the plant. In addition, recovery procedures should be carried out on a time scale to ensure that the break flow to the secondary system is terminated before the water level in the ruptured SG reaches the SG outlet nozzle.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.6.3.3 Regulatory Basis

The acceptance criteria for the SGTR event, a PA, are based on guidance from SRP section 15.0.3 and regulatory requirements which include:

- 10 CFR Part 50, Section 50.34(a)(1), "Contents of applications; technical information," as it relates to the evaluation and analysis of the offsite radiological consequences of postulated accidents with fission product release.

- GDC 19, 10 CFR Part 50, Appendix A, "Control room," as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.

- 10 CFR Part 100, Section 100.21, "Non-seismic siting criteria," as it relates to the evaluation and analysis of the radiological consequences of postulated accidents for the type of facility to be located at the site in support of evaluating the site atmospheric dispersion characteristics.

- 10 CFR Part 50, Appendix E, Paragraph IV.E.8, "Emergency Planning and Preparedness for Production and Utilization Facilities," as it relates to adequate provisions for an onsite technical support center (TSC) from which effective direction can be given and effective control can be exercised during an emergency.

MHI conservatively adopts two additional acceptance criteria: (1) to not allow SG overfill and (2) to maintain the RCS and main steam pressures below 110 percent of their respective design pressure to assure that rupture of the primary or steam system piping does not occur.

The section below describes the staff's evaluation of plant thermal-hydraulic response of the SGTR event, including the SG mass releases to be used for the evaluation of radiological consequences. The radiological consequence evaluation is described separately in Section 15.0.3.8 of this SER.

15.6.3.4 Technical Evaluation

Section 15.6.3 of the DCD describes the applicant's evaluation of a SGTR event occurring at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods.

The SGTR event is assumed to be a double-ended rupture of a single SG tube at the tubesheet on the cold end of the tube. A sensitivity study was provided in Appendix F of US-APWR topical report MUAP-07010-P, which compares the break flow rates for the breaks located at the cold side and hot sides of the tube sheet, and at the top of the U-bend. The result shows that a double-ended break at the cold side has the highest break flow rate. Therefore, this break location assumption is acceptable because it maximizes total primary-to-secondary leakage for both choked and unchoked conditions.

Upon initiation of an SGTR, low pressurizer pressure and low pressurizer level alarms are actuated. The flow from the charging pumps of the CVCS increases, and the pressurizer heaters are actuated in an attempt to maintain pressurizer level and RCS pressure. The continuous loss of reactor coolant inventory leads to a reactor trip on low pressurizer pressure. Automatic reactor trip can also be actuated by overtemperature ΔT, high-high SG water level, or ECCS actuation. Plant cooldown after the reactor trip leads to a rapid decrease of both RCS pressure and pressurizer level, which is balanced by charging flow and operation of the pressurizer heaters. Assuming a coincident loss of offsite power, steam is released through the MSRV to the atmosphere.

DCD Section 15.0.0.5 states that nonsafety-related systems are not required to mitigate the consequences of events, and only safety-related systems are credited in the safety analyses. The engineered safety features available for mitigation of the SGTR event include the EFWS automatic actuation, EFW isolation, and the ECCS. In response to RAI 808-5921, Question 15.06.03-8, the applicant provided a list of the systems, components, and instrumentation that are credited for mitigation in the safety analysis of the SGTR. All of the primary systems and components credited for the mitigation of the event are safety grade. The applicant also identified that the high sensitivity main steam line radiation (N-16) alarms (two channels per loop) are available, and the radiation monitors in the main steam line, condenser vacuum pump exhaust line, and the SG blowdown water are nonsafety-grade instrumentation. These

instrumentations are used to detect the SGTR event and identify the ruptured SG. However, an SGTR event will also actuate the low pressurizer pressure and low pressurizer level alarms. In addition, there are safety-related pressurizer water level and SG water level indications that can be used as backup for the detection of an SGTR event and identification of the ruptured SG. Therefore, the staff finds that the nonsafety-related radiation monitors provide redundant indications and are acceptable.

The mitigation of an SGTR relies heavily on timely operator actions to stabilize the plant and to terminate the primary-to-secondary leakage, thereby limiting release of contamination to the atmosphere and preventing the water level in the ruptured SG from reaching the SG outlet nozzle. The operator actions include recognizing the occurrence of an SGTR and tripping the reactor (if not automatically tripped already); identifying and isolating the ruptured SG; opening the MSDV on the intact SGs to reduce the RCS temperatures; opening the pressurizer safety depressurization valve (SDV) and stopping the safety injection flow to reduce the RCS pressure to equalize with the secondary pressure. These operator actions assumed in the SGTR analysis will be discussed later in the section.

The SGTR events are analyzed using the MARVEL-M computer code to determine the flow through the ruptured tube into the affected SG secondary system and subsequent releases of fluid to the environment. The MARVEL-M code, which is described in topical report MUAP-07010-P, has been reviewed by the staff for simulating the SGTR event. The evaluation of MARVEL-M also included an independent analysis of the SGTR event using both the RELAP5/MOD3.3 and MARVEL-M codes. It provided independent verification that analysis conservatisms claimed by the applicant were indeed conservative. As described in the SER on MUAP-07010-P, the staff concluded that MARVEL-M is acceptable for the SGTR analysis.

The applicant evaluated two SGTR cases: (1) a radiological dose evaluation (RDE) case and (2) an SG margin-to-overfill (MTO) case. The RDE case calculates the maximum steam release to the atmosphere via secondary system that provides input for the radiological dose calculation. One of the concerns related to an SGTR is the possibility of overfill of the SG secondary side, resulting in the accumulation of water in the steam line to challenge its structural integrity. The MTO case evaluates the margin to SG overfill. The following assumptions are made for both the RDE and the MTO cases:

- Initial power level at 102 percent of the rated thermal power.
- A coincident loss of offsite power at the time of reactor trip.
- The pressurizer pressure assumed to be 30 psi above the nominal value.
- The EFW flow rate supplied to each intact SG assumed to be at the minimum flow rate until EFW isolation.
- Minimum moderator density coefficient and maximum Doppler power coefficient shown in Figure 15.0-2.
- The reactor trip simulation with the trip time delay specified in Table 15.0-4, and the RCCA insertion and the scram reactivity depicted in Figures 15.0-3 “RCCA Displacement versus Time following Reactor Trip,” and 15.0-4 “Negative Reactivity versus Time following Reactor Trip,” respectively, of the DCD.
- Pressurizer heater and CVCS available.

The staff found these assumptions are suitably conservative and acceptable.

The limiting single failure for both cases is assumed to be the failure of one of the four EFWS trains, which results in one of the remaining SGs not receiving EFW flow. In response to RAI 808-5921, Question 15.06.03-4, the applicant provided a systematic evaluation of the system and component failures for the determination of the limiting single failure for the SGTR for both cases. The staff has reviewed this evaluation and agreed that the loss of one EFWS train, which reduces the heat removal capability, is the most limiting.

A few assumptions are different for the RDE and the MTO evaluations. For example, the initial reactor coolant temperature is assumed to be 4°F above the nominal value for the RDE, but 4°F below the nominal value for the MTO evaluation. The applicant stated, in the response to RAI 808-5921, Question 15.06.03-5, that a higher initial RCS temperature would result in higher temperature of the primary-to-secondary leakage, which is easier to vaporize and ultimately result in an increased amount of vapor released from the secondary side. This is a conservative assumption for the RDE case. On the other hand, a lower initial RCS temperature would reduce the amount of vaporized leakage and result in an increase in water level inside the SG. This is more conservative for the MTO evaluation. Therefore, these assumptions are acceptable.

The MFW control system is assumed to be available for the RDE, but is not credited in the MTO evaluation. Since the ruptured SG water level is increasing during the SGTR event, the MFW control system would automatically reduce the feedwater flow. Hence, automatic MFW control results in a lower SG water level, which is not conservative for the SG MTO evaluation. Therefore, the MFW control system is not assumed for the MTO case. On the other hand, assuming the MFW control system available keeps the SG water level lower. This prevents the level from reaching the high-high SG water level reactor trip setpoint and delays the time of reactor trip. This delay time results in additional primary-to-secondary leakage, which is conservative for the RDE case.

The EFW supply to the ruptured SG is assumed to be the minimum and maximum values, respectively, for the RDE and MTO evaluations. A minimum EFW supply to the ruptured SG would result in higher SG secondary water temperature and an increased amount of vapor released from the secondary side, and is therefore a conservative assumption for the RDE case. On the other hand, a maximum EFW supply increases the secondary water level and is a conservative assumption for the MTO evaluation.

For the RDE, a stuck-open MSRV on the ruptured SG is conservatively assumed when the MSDVs on the intact SGs are opened, requiring subsequent operator action to isolate the release to the environment by closing the block valve. The assumption of a stuck-open MSRV on the ruptured SG in the RDE case causes an uncontrolled depressurization of the ruptured SG, increasing the primary-to-secondary pressure difference, and thus increases primary-to-secondary leakage and mass release to the atmosphere. On the other hand, the mass release through the MSRV would result in a lower SG water level, which is non-conservative, and therefore, no credit of MSRV is taken for the MTO evaluation.

The initial water level in the SG is assumed to be the nominal programmed level with positive and negative uncertainties, respectively, applied for the MTO and RDE cases, respectively. The initial SG water level is one of the key parameters affecting the results of the SG MTO analysis during an SGTR event. In response to RAI 808-5921, Question 15.06.03-7, the applicant provided the value of nominal programmed level and the uncertainty values applied. The uncertainty value applied to the initial SG water level includes the instrument uncertainty and additional margin. The initial water level is assumed to be the nominal programmed water level plus uncertainty for the ruptured SG, and the nominal programmed value minus uncertainty for

the intact SG in the MTO analysis. For the MTO evaluation, the ruptured SG uncertainty is added to reduce the margin to overfill. The intact SG uncertainty is applied in the negative direction to conservatively reduce the heat removal capability of the intact SG. The staff finds this to be conservative and acceptable. For the RDE case, the lower initial SG water level because of the application of negative uncertainty would result in larger vapor release and is therefore conservative.

The results of the analyses of both the RDE and MTO cases are described in DCD Section 15.6.3.4.3. The RDE case results are shown in Figures 15.6.3-1 “RCS Pressure versus Time, SGTR, RDE Analysis” through 15.6.3-12 “SDV Flow Rate versus Time, RDE Analysis” and the event sequence Table 15.6.3-1 “Time Sequence of Events for SGTR, RDE Analysis.” The MTO case results are shown in Figures 15.6.3-13 “RCS Pressure versus Time, SGTR, MTO Analysis” through 15.6.3-21 “SDV Flow Rate versus Time, MTO Analysis” and the event sequence Table 15.6.3-2 “Time Sequence of Events for SGTR, MTO Analysis.” Because the mitigation of the SGTR relies heavily on the timeliness of operator action, the staff evaluates the key operator actions and their timing described in the sequence of events. For example, the following operator actions are assumed in the RDE case:

- Manual reactor trip and MFW isolation at 15 minutes
- Identification and isolation (MSIV closure) of the ruptured SG at 20 minutes
- Opening of intact SG main steam depressurization valves (MSDV) at 25 minutes
- Opening of pressurizer safety depressurization valve (SDV) at ~45.3 minutes
- Closure of SDV at ~47.5 minutes
- Manual termination of ECCS at 48 minutes

Regarding the isolation of the ruptured SG, Section 15.6.3.4.3.c states that the MSIV is closed 1200 seconds after SGTR initiation and, therefore, EFW flow is not provided for the ruptured SG since the MSIV closed before the EFW initiated. In RAI 808-5921, Question 15.06.03-6, the staff requested the applicant to clarify whether the MSIV closure function initiates the isolation of the EFW flow to the ruptured SG. In its response, the applicant stated that the isolation of the EFW in the safety analysis is credited as an operator action and not an automatic isolation function. The applicant proposed to modify the DCD to state that EFW flow to the ruptured SG is also isolated by operator action when the MSIV is closed. The staff finds the proposed DCD modification acceptable and identifies **Confirmatory Item 15.06.03-1** to track the DCD modification.

In RAI 808-5921, Question 15.06.03-3, the staff requested that the applicant provide an evaluation of the operator actions and completion times credited in the SGTR analysis, consistent with the corresponding steps in the ERG. In its response, MHI indicated that MHI is currently developing the US-APWR ERG for the purpose of supporting plant-specific EOPs, and the ERG was expected to be completed by the end of December 2011. MHI also provided a table listing the manual operator actions assumed in the SGTR analysis. For each action, the corresponding step that performs this function in the ERG and the alarms and/or indications that are used to assist the operator in performing the step are described. The table shows the time available (i.e., the operator action completion time assumed in the safety analysis) and time required (the amount of time the operator would take to complete the action) to perform each action. In each case, the time required has margin to the time available. Therefore, the manual action completion times assumed in the safety analysis are acceptable. The staff notes that in DCD Tier 1, ITAAC Table 2.9-1, Item 10 specifies the design commitment that requires the V&V

program be conducted in accordance with the requirements of the V&V Program Implementation Plan. The operator action completion times assumed in the safety analysis will be verified through integrated system validation, described in DCD Tier 2 subsection 18.10.2.3, as part of the V&V Program. The integrated system validation is conducted using actual dynamic human-system interface (HSI) with high fidelity plant model simulation of the operational conditions samples, such as the SGTR event. Using a plant-specific simulator and its typical control room staff, the COL applicant will demonstrate the operator actions and completion times are consistent with those assumed in the design basis analysis.

For the RDE case, the results demonstrate that the reactor trip system and the ESFs, in conjunction with operator actions, can terminate the primary-to-secondary break flow and stabilize the RCS in a safe condition. The RCS pressure decreases during the event progression. The maximum ruptured SG pressure is less than 110 percent of the SG design pressure of 8274 kPa [1200 psia]. The resulting primary-to-secondary break flow rate, and the intact and ruptured SG atmospheric mass release rate shown in Table 15.6.3-3 “SGTR – Mass Releases Results” are used as inputs for the radiological calculations, which are discussed in Section 15.0.3.8 of this SER.
For the SG MTO case, the result shows that the water volume in the ruptured SG is less than the total SG volume when the break flow stops. Thus there is still margin to SG overfill. The RCS pressure decreases during the event progression, and the maximum SG pressure is less than 110 percent of the SG design pressure of 8274 kPa [1200 psia].

15.6.3.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.6.3.6 Conclusions

The staff has reviewed the SGTR analyses for both the RDE case and the MTO case described in DCD Section 15.6.3. The analyses were performed with the MARVEL-M code, which is approved by the NRC as discussed in Section 15.0.2 of this SER. The staff has reviewed the assumptions, parameters, and initial conditions in the accident analyses for a SGTR and, upon incorporation of the confirmatory item discussed in this section, concludes that they are conservative assumptions. The results show that there is still margin to SG overfill, and the RCS and the SG pressures are below the 110 percent of the design pressures. The RDE case provides the primary-to-secondary break flow rate, and the intact and ruptured SG atmospheric mass release rates to be used as inputs for the radiological calculations, which are described in Section 15.0.3.8 of this SER. As a result of the confirmatory item, the staff is unable to finalize its conclusion on Section 15.6.3 in accordance with requirements of 10 CFR Part 50 (Appendix A and Appendix E) and 10 CFR Part 100.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within Reactor Coolant Pressure Boundary

15.6.5.1 Introduction

This section documents the NRC staff evaluation of the applicant's DCD Tier 2 analyses of the US-APWR response to postulated LOCAs, including long-term cooling. These analyses were used to determine if the US-APWR complied with the requirements of the regulations for ECCS given in 10 CFR 50.46 and Appendix K to 10 CFR 50 and the applicable general design requirements discussed in Section 6.3 of NUREG-0800.

LOCAs are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system, from piping breaks in the RCPB. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the RCPB. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The buildup of boric acid due to coolant vaporization, if left uncontrolled, could reach precipitation limits and block the coolant channels in the core, preventing adequate heat removal for any size break.

GDC 35 requires each PWR and BWR to be equipped with an ECCS that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS given in 10 CFR 50.46 and Appendix K to 10 CFR 50 and the applicable general design requirements discussed in Section 6.3 of NUREG-0800.

The review of the applicant's analysis of the spectrum of postulated LOCAs was closely associated with the staff's review of the ECCS, as described in Section 6.3 of NUREG-0800. The staff evaluated whether the entire break spectrum (break size, location and orientation) was addressed; whether the appropriate break locations, break orientations, break sizes, and initial conditions were selected in a manner that conservatively predicted the consequences of the LOCA for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance was provided. For postulated break sizes and locations, the review included: (1) the postulated initial reactor core and reactor system conditions, (2) the postulated sequence of events including time delays prior to and after emergency power actuation, (3) the calculation of the power, pressure, flow and temperature transients, (4) the functional and operational characteristics of the reactor protective, and (5) ECCS systems in terms of how they affected the sequence of events, and operator actions required to mitigate the consequences of the accident.

The staff also considered post-LOCA long-term cooling to assure that the applicant identified the operator actions necessary to successfully control and prevent boric acid precipitation. Analyses of both large break and small break LOCAs were evaluated by the staff to identify the timing for boric acid precipitation.

The timing for the switch to simultaneous injection for large breaks, switching one Safety Injection (SI) pump from direct vessel injection (DVI) to hot leg injection, was evaluated to

assure that the timing was determined using acceptable analysis methods. A spectrum of small breaks was analyzed by the applicant to identify other means to control boric acid precipitation when RCS pressure remains too high to enable flushing of the core through a simultaneous injection lineup during the long term. The staff reviewed all equipment and operator action times to determine whether they were clearly identified in the analyses.

Confirmatory calculations were performed by the staff to assure that modeling techniques used by the applicant were conservative and that the causes for differences in results of the applicant's analysis compared to those for similar plants were understood.

15.6.5.2 Summary of Application

DCD Tier 1: There were no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant provided a DCD Tier 2 description in Section 15.6.5 [References 15-1, 15-2 and 15-46], summarized here as follows:

The application explained the LOCA PAs that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system. The coolant loss occurs from piping breaks in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS.

Various size breaks were examined to determine the conditions of the RCS, reactor core, and containment vessel, and to demonstrate that the ECCS had the capability to mitigate each LOCA. For the US-APWR, the spectrum of breaks was categorized under large break and small break LOCAs for the purpose of reporting bounding results. A large break was defined as a break with a total cross-sectional area equal to or greater than 1.0 ft^2. A small break was defined as a piping break within the RCPB with a total cross-sectional area up to 1.0 ft^2.

The small break LOCA spectrum considered breaks large enough that the CVCS charging pumps could not provide sufficient makeup water to the RCS, and ECCS would be actuated. For very small breaks where the charging pumps have the capability to make up the leakage, the pressurizer level and pressure would be sustained and the ECCS would not be actuated.

In the transient and accident analyses for the US-APWR, both large break and small break LOCAs were classified as postulated accidents. They are not expected to occur during the life of the plant, but postulated as a conservative design basis.

ITAAC: The ITAAC related to this area of review are identified in Section 15.0.0.2 of this SER.

Technical Specifications: The TS related to this area of review are identified in Section 15.0.0.2 of this SER.

15.6.5.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (hereafter referred to as NUREG-0800 or the SRP), Section 15.6.5, “Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary,” and are

summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 15.6.5.

1. 10 CFR 50.46, as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the RCPB.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 35, as it relates to demonstrating that the ECCS would provide abundant ECC to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that: (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

   The analyses should reflect that the ECCS has suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities available, such that the safety functions could be accomplished assuming a single failure. In addition, consideration should be given to the availability of onsite power (assuming offsite electric power is not available with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available).

4. 10 CFR 100 or 10 CFR 50.67, as they relate to mitigating the radiological consequences of an accident.

Acceptance criteria adequate to meet the above requirements include:

1. Compliance with 10 CFR 50.46 [Reference 15-3] requires that light water cooled nuclear power reactors be equipped with an ECCS designed so that core performance following postulated LOCAs conforms to specified criteria related to limiting core damage. RG 1.157, “Best Estimate Calculations of Emergency Core Cooling System Performance,” and Appendix K to 10 CFR 50 [Reference 15-4], provide guidance and requirements on evaluation models needed to demonstrate compliance with the acceptance criteria. Appendix K also specifies documentation required for evaluation models.

   The requirements specified in 10 CFR 50.46 provide an acceptable and conservative means of calculation of the consequences of LOCAs from a spectrum of pipe break sizes and locations that have been subject to careful review and experimental verification.

   If the calculations of the performance of the ECCS are conducted in accordance with these methods, there is a high level of probability that the acceptance criteria on core performance will not be exceeded and damage to the core and offsite consequences will be minimized.

Meeting the requirements outlined in the references provides assurance that following a LOCA, the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR 100 or 10 CFR 50.67.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values for the relevant monitored parameters and instrument indications.

3. Compliance with GDC 35 requires that a means of providing abundant ECC be provided that will transfer heat from the reactor core in the event of a LOCA, and that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure. GDC 35 specifies that an ECCS be installed in all nuclear power reactors. Section 15.6.5 of NUREG-0800 specifies the analytical procedures that are to be followed to establish that the ECCS will function to meet acceptance criteria specified in 10 CFR 50.

   Meeting the requirements of GDC 35 will provide assurance that following a LOCA the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR 100 or 10 CFR 50.67.

4. 10 CFR 100 and 10 CFR 50.67, Reactor Site Criteria, describe criteria that guide the Commission in its evaluation of the suitability of proposed sites for nuclear power and test reactors. 10 CFR 100 or 10 CFR 50.67 specify radiation dose guidelines that should not be exceeded in the event of PAs including LOCAs.

   In order to satisfy the requirements of 10 CFR 100 or 10 CFR 50.67, the applicant must demonstrate that the offsite doses resulting from various accidents presented in the safety analysis report (SAR) are within the guideline values. Meeting the guideline doses is achieved by a combination of engineered safety features installed in the nuclear facility, an effective ECCS, and locating the nuclear plant in an area that does not exceed population density requirements.

   Meeting the nuclear power plant siting criteria provides a level of assurance that the plant will pose no undue risk to the public as a result of the consequences of LOCAs.

15.6.5.4 Technical Evaluation

15.6.5.4.1 Large-Break LOCA

The large-break LOCA analysis was done with a best estimate evaluation methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) developed by Westinghouse (WCAP-16009-P-A). This methodology uses the WCOBRA/TRAC and HOTSPOT computer codes (WCAP-12945-P-A and WCAP-16009-P-A) to simulate 124 LBLOCA cases in which relevant parameters are randomly varied. The relevant parameters are determined ahead of time by constructing a PIRT (Phenomena Identification and Ranking

Table). The application of the ASTRUM methodology and the US-APWR PIRT process are described in Topical Report MUAP-07011-P, "Large Break LOCA Code Applicability Report for US-APWR" [Reference 15-5]. The staff's evaluation of the methodology's application is given in the SE [Reference 15-6] to that Topical Report.

In RAI 352-2369, Question 15.6.5-2, the staff asked how the least favorable power shape was determined for each break size. The response to RAI 352-2369, Question 15.6.5-2 [Reference 15-7] clarified how core power shapes are selected in the applicant's best-estimate LBLOCA methodology. The power shape is one of the statistical parameters in the methodology. The axial power distribution for each case was randomly sampled from the operating range given in Figure 15.6.5-8, "Axial Power Shape Operating Space Envelope for Large Break LOCA," of the DCD. The response is acceptable because it clarifies that choosing a least favorable power shape for each break is not part of the methodology.

In RAI 352-2369, Questions 15.6.5-3 and 15.6.5-4, the staff sought assurance that the High Head Safety Injection (HHSI) flow curves were modeled in a conservative fashion. In its response to questions 15.6.5-3 and 15.6.5-4 [Reference 15-7], the applicant noted that HHSI flow characteristics for both minimum and maximum safeguards (Figures 6.3-15, "High Head Safety Injection Flow Characteristic Curve - Minimum Safeguards," and 6.3-16, "High Head Safety Injection Flow Characteristic Curve - Maximum Safeguards," of the DCD) are based on conservative assumptions and account for the head loss due to the accumulation of debris on the ECC/CS strainer. The head loss is discussed in MUAP-08001(R5), "US-APWR Sump Strainer Performance," Revision 5. The acceptance of the applicant's assumed ECC/CS strainer head loss has a direct bearing on the conservativeness of the ECC flow curve being used for the LOCA analyses. **An Open Item 15.6.5-1 has been created in this SE as the staff has not yet completed its review of MUAP-08001.**

The idea behind partitioning the LBLOCA transient into phases was based on a change in some dominant phenomenon. In RAI 352-2369, Question 15.6.5-5, the staff asked what dominant phenomena separates the blowdown and refill phases of the LBLOCA in the applicant's analysis. The response to Question 15.6.5-5 [Reference 15-7] explained that the dominant phenomenon that is changing during the transition from the blowdown phase to the refill phase of the LBLOCA is ECC bypass. The refill phase begins when ECC bypass ends. The staff concurs that this response provides a reasonable definition of the boundary between the blowdown and refill phases; the response is therefore acceptable.

Secondary side pressure is a major influence on primary/secondary heat transfer during a LBLOCA. In order to confirm that secondary pressure was being properly treated, the staff requested a plot of the secondary side pressure response in RAI 352-2369, Question 15.6.5-6. In response to Question 15.6.5-6 [Reference 15-7], the applicant provided plots of the secondary side pressure in each of the four SGs for the LBLOCA Reference Case. The plots show that all four SG pressures behaved similarly. The pressure rose rapidly when SG isolation occurred early in the transient, and then declined due to reverse heat transfer across the SG tubes. The plots show that the calculated secondary side pressures are reasonable; the response is therefore acceptable.

The accumulator coolant temperature is set to 35 C (95 °F) in the LBLOCA Reference Case, but is a sampled parameter in the ASTRUM analysis. In RAI 352-2369, Question 15.6.5-7, the staff asked how the interaction of cold ECCS fluid with the saturated steam in the cold legs is modeled. The response to Question 15.6.5-7 [Reference 15-7] noted that the effect of the condensation of vapor due to accumulator injection is considered in the interfacial heat and

mass transfer models in WCOBRA/TRAC (M1.0). The response also noted that the accumulator's non-condensable cover gas does not enter the primary system until after the core is quenched. The response demonstrates that condensation due to accumulator injection is being treated in the analysis as is the effect of non-condensable gas injection. The response is therefore acceptable.

HOTSPOT is used in the applicant's best estimate LBLOCA methodology to compute the thermal response of the hot rod. RAI 352-2369, Question 15.6.5-8 asked the applicant to provide a description of HOTSPOT, including how it interfaces with WCOBRA/TRAC (M1.0). The response to Question 15.6.5-8 explained that HOTSPOT calculates the effect of uncertainties at a single axial location of the fuel rod. It simulates the following phenomena: transient heat conduction in the fuel pellet and cladding, cladding burst and strain, inside and outside cladding oxidation, and fuel relocation following cladding burst. HOTSPOT interfaces with WCOBRA/TRAC in one direction. WCOBRA/TRAC writes a binary file containing requisite transient data (phasic temperatures and heat transfer coefficients) at each axial location. One of the ASTRUM scripts reads this binary file and sets up the necessary boundary conditions for the HOTSPOT calculation. The response to Question 15.6.5-8 [Reference 15-7] is acceptable because it has provided the requested description of HOTSPOT and the interface between it and WCOBRA/TRAC.

DCD Tier 2, Table 15.6.5-1, "US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis," contains the major plant parameters used in the ASTRUM analysis. The staff issued RAI 352-2369, Question 15.6.5-9 requesting a description of uncertainty parameters related to models in the computer codes. The response to Question 15.6.5-9 presented the uncertainty parameters related to the models in the computer codes comprising the best-estimate LBLOCA methodology. These parameters were not given in the DCD because they are considered proprietary. The applicant noted that the bases and probability density function (PDF) for most of the model parameters are given in WCAP-16009-P-A. The assumed accumulator bounding loss coefficients bias and uncertainty values are given in MUAP-07011-P (R1), "Large Break LOCA Code Applicability Report for US-APWR." The response to Question 15.6.5-9 is acceptable because it presented all of the methodology's statistical parameters and provided justification for the PDFs of each parameter in DCD Table 15.6.5-1. The staff evaluated the methodology's statistical parameters and found them acceptable in its SE for MUAP-07011-P (R3).

Section 15.6.5.3.3.1 provides plots of several thermal-hydraulic parameters for the LBLOCA Reference case, but provides only the cladding temperature response for the ASTRUM limiting cases. The response to RAI 352-2369, Question 15.6.5-10 [Reference 15-7] addressed the staff's request for additional information for those cases by providing plots of pressure, integrated break flow, and hot assembly flow. The response also addressed the staff's request for an explanation of the oscillations of the hot assembly exit flow rate (Figure 15.6.5-2, "Hot Assembly Exit Vapor, Entrainment, Liquid Flow Rates for Large Break LOCA - Reference Case"). The first part of the applicant's response is acceptable because it provided the requested figures. The response to the second part, which states that the oscillations appear to be related to the unstable nature of churn-turbulent flow, is unsatisfactory. However, the core flow oscillations are addressed in the staff's evaluation [Reference 15-6] of MUAP-07011-P (R3), "Large Break LOCA Code Applicability Report of US-APWR." That evaluation found that the core flow oscillations had little impact upon computed PCT and were therefore acceptable. Based upon this finding, the response to the second part of Question 15.6.5-10 is no longer needed.

In RAI 352-2369, Question 15.6.5-11, the staff asked whether the PCTs in Figure 15.6.5-9, "HOTSPOT PCT versus Effective Break Area Scatter Plot for Large Break LOCA," were blowdown or reflood PCTs. In response to Question 15.6.5-11 [Reference 15-7], the applicant explained that Figure 15.6.5-9 shows the PCTs for all 124 ASTRUM cases. A specific case's PCT may occur during the blowdown period or the reflood period depending upon the values of the statistical parameters assigned to that case. The case with the maximum PCT had a break discharge coefficient, multiplied by break flow area, nearly equal to the cold leg flow area. Cases which had larger break areas had lower PCTs because the tendency of PCT to increase with break area was offset by the influence of other statistical parameters. The response adequately explains what is being plotted in Figure 15.6.5-9 and why the PCT from the ASTRUM analysis does not monotonically increase with break area; it is therefore acceptable.

The limiting core wide oxidation (CWO) case is usually the case which keeps the cladding temperature high for the longest period of time. This case may or may not be the case with the highest PCT or local maximum oxidation (LMO). RAI 352-2369, Question 15.6.5-12, asked if the cladding temperature plots, Figures 15.6.5-10, "HOTSPOT Cladding Temperature Transient at the Limiting Elevation for the PCT Limiting Case for Large Break LOCA," 15.6.5-11, "HOTSPOT Cladding Temperature Transient at the Limiting Elevation for the LMO Limiting Case for Large Break LOCA," and 15.6.5-12, "PCT Transient for the CWO Limiting Case for Large Break LOCA," in the DCD were from a single run. The response to Question 15.6.5-12 explained that Figures 15.6.5-10, 15.6.5-11, and 15.6.5-12 of the DCD are the cladding temperature responses for three different ASTRUM cases, the limiting PCT, LMO, and CWO cases, respectively. The value of CWO is selected as the most limiting oxidation value for the rod within the hot-assembly. Therefore, the case which keeps the cladding temperature higher over the transient tends to be the CWO limiting case. The cladding temperature responses differ between DCD revisions because the LBLOCA analysis was redone in each DCD revision. The response provides the clarification sought and is therefore acceptable.

In order to verify the statistics and confirm the highest values of PCT, LMO, and CWO, the staff asked the applicant to provide the cumulative distributions for these parameters. The response to RAI 352-2369, Question 15.6.5-13 [Reference 15-7], provided the requested cumulative distribution functions for the PCT, LMO, and CWO for all of the 124 ASTRUM cases. The response allowed the staff to verify the highest values for the three safety parameters. Therefore, the response is acceptable.

The staff requested an explanation of the basis for the minimum containment pressure assumed in the LBLOCA analysis in RAI 352-2369, Question 15.6.5-14. The applicant's response [Reference 15-7] was that the minimum containment pressure corresponds to the pressure obtained in DCD Tier 2 subsection 6.2.1.5, "Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System." The response is acceptable to the staff because it provides the technical basis for the containment pressure boundary condition. The acceptability of the minimum containment pressure calculation is evaluated in the staff's review of MUAP-07011-P (R0), "Large Break LOCA Code Applicability Report of US-APWR."

RAI 352-2369, Question 15.6.5-15, asked if there had been any changes in the LBLOCA evaluation model (MUAP-07011 (R0)) due to the staff's review. The applicant responded that there had been no changes. This response was appropriate at the time the response was given (July 2009). Subsequently, the LBLOCA analysis was revised. For Revision 2 of the DCD, the LBLOCA analysis was redone due to modifications to WCOBRA/TRAC and HOTSPOT. The modifications were evaluated by the staff as part of its SE for Topical Report MUAP-07011-P.

For Revision 3 of the DCD, the LBLOCA analysis was redone again because of a re-analysis of the minimum containment pressure curve, modifications to WCOBRA/TRAC and HOTSPOT, changes in the uncertainty ranges for core power and SI fluid temperatures, and the addition of a bounding bias to accumulator flow. The staff's evaluation of the minimum containment pressure curve is contained in the SER for DCD Tier 2, Subsection 6.2.1.5. The staff's evaluation of the uncertainty ranges for core power and SI fluid temperatures is given in the Topical Report MUAP-07011-P SE [Reference 15-6].

Open Item 15.6.5-2 has been created in this SE to verify that the accumulator flow rate bias used in the LBLOCA evaluation is conservative relative to that determined in the Advanced Accumulator Topical Report, MUAP-07001 (Reference 15-8) which is still under staff review. The current (DCD Tier 2 Revision 3, [Reference 15-46]) LBLOCA analysis has been reviewed and found acceptable by the staff.

15.6.5.4.2 Small Break LOCA

The applicant performed SBLOCA analyses using the M-RELAP5 [Reference 15-32] computer code (Version 1.6). M-RELAP5 is a one-dimensional, two-fluid computer code used to model flow of a two-phase steam-water mixture in a nuclear reactor system under transient conditions. The M-RELAP5 code was based on RELAP5-3D [Reference 15-10]. Although the code has multi-dimensional thermal-hydraulics modeling capability, all analyses used only one-dimensional models. The code models non-equilibrium thermodynamics; has specialized models for phenomena such as choked flow, counter-current flow limit (CCFL), critical heat flux (CHF) and pump performance; and contains models for analyzing conductive and convective heat transfer in solid structures during a transient event.

The staff's evaluation and acceptance of the M-RELAP5 code for US-APWR SBLOCA analysis is discussed in Reference [Reference 15-11]. The staff concluded that the M-RELAP5 code, when applied using the methodology documented in MUAP-07013-P, "Small Break LOCA Methodology for US-APWR" [Reference 15-32], is acceptable for performing SBLOCA analysis for the US-APWR to demonstrate compliance with the requirements of 10 CFR 50, Appendix K.

MHI Sensitivity Analyses

MUAP-07013-P, "Small Break LOCA Methodology for US-APWR," was supplemented with a Technical Report on Small Break LOCA Sensitivity Analyses for US-APWR [Reference 15-12, 15-13, 15-14, and 15-47] submitted in support of the US-APWR Design Certification Application [Reference 15-12, 15-13]. To establish the limiting small break location, the applicant analyzed a spectrum of small break LOCAs as a part of these sensitivity studies. Sensitivity studies were run with breaks of 2-inch and 1-ft^2 in the cold leg, hot leg, and crossover leg, and with double ended breaks in the DVI line and in the pressurizer steam phase. Breaks in the cold leg were determined to be limiting.

Subsequently, a break spectrum analysis was performed with integer inch diameter break sizes from 1-inch through 13-inch and 1-ft^2 (13.5-inch) in the cold leg. Additional breaks were analyzed with 0.5-inch intervals in break diameter to determine the final limiting break size resulting in the highest PCT. Two distinct PCT peaks were identified, one during the loop-seal clearance phase, for the 7.5 inch break, and one later in time during the boiloff, for the 1.0 ft^2 break.

The results of the break spectrum sensitivity calculations identified the limiting break conditions including break location, break size, and break orientation. Breaks in the bottom of the cold leg piping were determined to be limiting for PCT. During the loop-seal clearing phase, a PCT of 761 °F (405 °C) was calculated for the 7.5-inch cold leg bottom break, while during the boil-off phase a PCT of 1328 ºF (720 °C) was calculated for the 1.0 ft^2 cold leg bottom break. A PCT of 789 °F (420 °C) was calculated for the DVI line break.

At these PCT values, the local calculated clad oxidation, and therefore the average core-wide clad oxidation, was minimal (less than 0.2 percent) and the subsequent discussions focus on PCT as the limiting criteria.

Other sensitivity studies performed by the applicant included: (1) noding near the break point, (2) noding near the DVI injection point, (3) time step size, (4) noding of SG tubes and crossover leg (loop seal), (5) no single failure assumption, and (6) offsite power available. The time step sizes used were shown to be sufficiently small to assure code solution convergence. The analyses results also demonstrated that the assumptions for single failures (one ECCS train disabled) and for loss of offsite power were satisfactorily selected as the limiting cases. The staff reviewed these additional sensitivity studies and determined that they are sufficient to meet the requirements of 10 CFR 50 Appendix K sections I.C.1a, I.C.1d, II.2 and II.3.

Requests for Additional Information

As a part of the review of Section 15.6.5 of the US-APWR Design Certification Application [References 15-12 and 15-13] the staff requested additional information related to the SBLOCA modeling approach and the results of the calculations. Three requests for information RAI 352-2369, 514-4040 and 513-4170 were made in conjunction with the review of the SBLOCA analysis in DCD Tier 2 Section 15.6.5 and the sensitivity analyses in MUAP-07025-P, "Small Break LOCA Sensitivity Analyses for US-APWR." Two requests for additional information [References 15-18 and 15-19] related to the confirmatory analyses [Reference 15-20] performed by the staff were made.

The applicant responded to the first request for additional information in letter UAP-HF-09384 [Reference 15-7]. The applicant responded to the second request for additional information in letter UAP-HF-10038 [Reference 15-21]. The applicant responded to the third request for additional information in letter UAP-HF-10039 [Reference 15-22] and letter UAP-HF-10042 [Reference 15-23]. The applicant responded to the first request for additional information related to the confirmatory calculations [Reference 15-18] in letter UAP-HF-09492 [Reference 15-24] and to the second request in letter UAP-HF- 09512 [Reference 15-25]. Supplemental responses dealing with SBLOCA RAI Question CA-5 [Reference 15-18] were provided in UAP-HF-10040 [Reference 15-26] an UAP-HF-10059 [Reference 15-27]. As a consequence of changes made to the M-RELAP5 code from version 1.5 to version 1.6, it was necessary for the applicant to revise the responses to several RAIs. The revised responses were documented in Reference 15-28.

Phases of the Event

Since the importance of the various phenomena differs as the small break LOCA event develops in time, the applicant divided the event into five phases: blowdown, natural circulation, loop seal clearance, boiloff, and core recovery. The duration of each phase depends on the break size and the performance of the ECCS. The defining characteristics of each phase were

described by the applicant in the submittal [Reference 15-2 and Reference 15-46] and are briefly summarized here for the 7.5 inch cold leg break.

Following initiation of the break, the RCS primary side rapidly depressurizes until flashing of the hot coolant begins. Reactor trip is initiated at the low pressurizer pressure setpoint of 1860 psia (12.82 MPa). Closure of the condenser steam dump valves isolates the SG secondary side. As a result, the SG secondary side pressure rises to the SG safety valve set point of 1296 psia, and steam is released through the safety valves for the 7.5 inch break case only.

The ECCS actuation signal is generated at the time the pressurizer pressure decreases to the low pressurizer pressure setpoint of 1760 psia (12.13 MPa), and safety injection initiates after a time delay. The RCPs automatically trip, after a 3 second delay, upon the ECCS actuation signal.

The rapid depressurization (blowdown phase) ends when the pressure falls to just above the saturation pressure of the SG secondary side, which is at the safety valve set point.
The break flow in the RCS is single-phase liquid throughout the blowdown period.

When the blowdown phase ends, two-phase natural circulation is established in the RCS loops. As more coolant is lost from the RCS through the break, steam accumulates in the downhill side of the SG tubes and in the crossover leg. The natural circulation phase ends when there is insufficient driving head on the cold leg side of the loops, due to the accumulation of steam between the top of the SG tubes and the loop seals.

The SG safety valve setpoint was not reached for the 1-ft^2 break. It was difficult to define the end of blowdown period for larger break sizes (larger than a 10-inch break for the US-APWR), because the primary pressure rapidly decreases below the secondary pressure and the natural circulation period and loop seal clearance periods were nonexistent. For these breaks, the end of the blowdown phase occurs when the primary inventory begins to increase from the accumulator injection.

In RAI 352-2369, Question 15.06.05-19, the staff requested the applicant to provide figures to show temporal changes in the flow rate over the natural circulation phase. The applicant provided figures showing the natural circulation flow rate for the 1-ft^2 and 7.5-inch cold leg break cases and for the DVI line break case [Reference 15-18]. These figures provided the information needed by the staff to confirm that the end of natural circulation phase had been correctly determined and the response to Question 15.06.05-19 is therefore acceptable.

Until the loop seals clear the break remains covered with water so the RCS water inventory continues to decrease and steam volume in the RCS increases. The relative pressure in the core increases, which, together with the loss of coolant inventory through the break, causes the liquid levels in the core and the SG to continue to decrease. If during this process, the core mixture level drops below the top of the core, the cladding will experience a dryout and the cladding temperature in the upper part of the core will begin to rise. A heatup can occur during this loop seal clearance phase of the event. When the liquid level in the downhill side of the SGs is depressed to the elevation of the loop seals, the seals clear and steam in the RCS is vented to the cold legs. Break flow changes from a low quality mixture to primarily steam. This relieves the back-pressure in the core and the core liquid level is restored to the cold leg elevation by flow from the downcomer.

To verify the above statement that "until the loop seals clear the break remains covered with water" the applicant provided [Reference 15-7], in response to RAI 352-2369, Question 15.6.5-20 , figures of the void fraction upstream of the break for the three small break LOCA cases, 7.5-inch and 1 ft^2 cold leg breaks and the DVI line break. The staff used these figures to verify that the break remains covered with liquid prior to loop seal clearing. Thus the response to Question 15.6.5-20 is acceptable. In RAI 352-2369, Question 15.6.5-21, the staff requested that the applicant explain the term 'relative pressure in the core' that was used by MHI in its description of the core behavior during the SBLOCA event.

In response to Question 15.6.5-21, the applicant explained [Reference 15-7] that 'relative pressure in the core' indicates the pressure difference between the top of the core and the top of the downcomer and that this pressure difference results from the hydrostatic pressure difference between the uphill and downhill sides of the SG U-tubes. MHI noted that the effect of the relative pressure difference is shown in Figure 15.6.5-20, "Core/Upper Plenum Collapsed Level for 7.5-inch Small Break LOCA" of the DCD. The response clarified the description in the DCD and is therefore acceptable.

After the loop seals clear, the RCS primary side pressure falls below that of the secondary side due to the increase of the break flow quality, resulting in a lower mass flowrate but a higher volumetric flow through the break. The vessel mixture level may decrease as a result of the core boiling in this phase. If the RCS pressure is too high for the injection system to make up for the boil-off rate, the core might uncover before the RCS depressurizes to the point where the SI pumps (and accumulator, when the RCS pressure drops to a sufficiently low value) deliver ECCS water to the RCS at a rate greater than the break flow. Fuel heatup can occur during this boiloff period. Later the pressure will drop and ECCS flow will be sufficient to recover and cool down the core in this recovery phase.

M-RELAP5 Plant Model

The US-APWR M-RELAP5 model was described in MUAP-07025-P (R3), "Small Break LOCA Sensitivity Analyses for US-APWR," [Reference 15-47]. The report provided details on the systems and components and how those systems were represented in the input model. Components in the primary system include the reactor vessel, the SG primary side, the reactor coolant pumps, the pressurizer, the main coolant pipe and pressurizer surge line, the accumulators, and direct vessel injection from the SI pumps. The secondary system includes the SG secondary side, main feedwater systems, main steam systems, emergency feedwater systems, and safety valves. Nodalization diagrams were provided for the US-APWR model and for individual components. The staff reviewed the M-RELAP5 model, including the assumed operating conditions and system performance. A number of RAIs were generated and responded to as a part of the plant model review.

RAI 352-2369, Question 15.06.05-23, requested that the applicant discuss the noding sensitivity studies for components other than those included in the sensitivity studies (including volume and junction options), especially for the heated region of the core and vessel. The applicant responded [Reference 15-7] that US-APWR nodalization for the core and vessel regions was finer than the acceptable nodalization specified in the RELAP5-3D users' guidelines for analyses of Westinghouse-designed PWRs [Reference 15-29]. The nodalization scheme for the US-APWR was the same as that used by the applicant in the integral test facility assessment cases. The staff verified that volume and junction options for the volumes and junctions followed the guidelines presented in Section 3.3.2 of the RELAP5-3 D User's Guidelines [Reference 15-29]. The explanations provided by the applicant justified the nodalization used in

the core and vessel, including volume and junction options; therefore, the response to Question 15.06.05-23 is acceptable.

In RAI 352-2369, Question 15.06.05-26, the staff requested the applicant to describe how the flow distribution in the core (fraction of flows through various bypasses modeled in the core) was validated. The applicant explained [References. 15-7 and 15-28] that a pressure loss coefficient for each bypass path was derived based on conventional experimental data and/or widely used correlations. The best estimate value of the bypass flow rate was calculated as 7.5 percent of the RCS flow rate.

The uncertainty regarding the core bypass flow was estimated by considering the manufacturing tolerances of the bypass flow paths and uncertainties in the pressure drop through the core and bypass flow paths. The total uncertainty of the bypass flow rate was conservatively estimated as 1.5 percent of the RCS flow rate (or 20 percent of the best estimate bypass flow).

Therefore, the maximum bypass flow rate was determined as 9 percent of the RCS flow rate. While there is uncertainty in the core bypass flow, the 20 percent assumed by the applicant is high compared to that used for existing plants, and therefore is acceptable because a high bypass flow is conservative. The core pressure drop used to determine the bypass flows was validated by a hydraulic test for the US-APWR fuel assembly completed in 2010 that provided confirmatory data. The applicant explained how the bypass flow was determined. Therefore, the response to Question 15.06.05-26 is acceptable.

In the response to RAI 352-2369, Question 15.06.05-27, the applicant discussed how the flow areas and friction factors were determined for the cross flow junctions in the vessel, and how the model and input data were validated [Reference 15-7]. The formula for calculating the flow area and the correlation for the pressure loss coefficient for the cross flow junction were provided. The modeling scheme was validated by the ROSA-IV/LSTF SBLOCA 5 percent cold leg break assessment, based on the comparison of the calculated cladding temperature in the hot assembly to the test data and the comparison of the calculated void fraction and liquid level for the hot assembly to the test data. The information provided in the applicant's response to Question 15.06.05-27 described how the cross flow areas and friction factors were determined. The response to Question 15.06.05-27 is acceptable.

In RAI 352-2369, Question 15.06.05-28, the staff noted that the nominal initial pressurizer level was assumed for the safety analyses. The applicant was asked to discuss the uncertainty of the level measurement and its impact on the pressure and the event progression (e.g., scram and ECCS initiation). The applicant responded [Reference 15-7] that, in the small break LOCA scenario, no trip signal is generated based on the pressurizer level. The initial level has a minor effect on the pressure response but this effect is offset by the change in inventory that is available to refill the primary system so that there is no significant effect on the calculated PCT. The staff agrees that the effect will be minor with increased level above nominal, resulting in a slower depressurization (conservative) due to the larger inventory of hot fluid available to flash, and is offset by the larger inventory available to refill the RCS (non-conservative). The applicant explained the effect of the uncertainty in initial pressurizer level. The response to Question 15.6-28 is acceptable. RAI 352-2369, Question 15.06.05-29 asked whether the approach used to determine the safety injection water temperature, from the in-containment RWSP, would result in a conservative boric acid concentration. The applicant responded [Reference 15-7] that the effects of boric acid on reactivity feedback are not modeled in M-RELAP5. The response to Question 15.06.05-29 is acceptable.

In RAI 352-2369, Question 15.06.05-30, the applicant was requested to discuss the uncertainty of the pressure measurement at the pressurizer and its impact on the SCRAM and ECCS timing. In the revised response [Reference 15-28], the applicant explained that the initial pressurizer pressure uncertainty of 30 psi (0.21 MPa) was added to the nominal value of 2250 psia (15.51 MPa). Setting the initial pressure at the maximum is conservative because it results in a maximum delay to the low pressure setpoint for reactor scram and ECCS initiation. The explanation provided by the applicant clarifies how the uncertainty in pressurizer pressure measurement was addressed in the analysis. The response to Question 15.06.05-30 is acceptable.

In RAI 352-2369, Question 15.06.05-31, the staff requested an explanation of how break orientation was modeled because the cold leg is modeled in one-dimension. In response to Question 15.06.05-31 [Reference 15-7], the applicant explained how M-RELAP5 allows for different break orientations (top, side, or bottom) and discussed validation studies regarding break orientation.

The crossflow junction model was used to simulate the break. This type of junction allows connections at cell faces that are perpendicular to the normal flow direction.

Also, the stratification entrainment/pull-through model was used to allow the user to specify that a junction is connected at the centerline, or on the top or bottom of a horizontal pipe. The model incorporates correlations that were developed for off-takes located at the top, bottom, and side of the horizontal pipe.

The model was assessed against the data used in its derivation, as well as by the simulation of LOFT SBLOCA experiment LP-SB-02. This explanation describes how break orientation was modeled. The response to Question 15.06.05-31 is acceptable.

In response to RAI 352-2369, Question 15.06.05-32, which requested the applicant provide a list of the heat structures included in the M-RELAP5 model, the applicant provided the list of the several regions for which heat structures were included for SBLOCA analysis [Reference 15-7]. The staff compared the list to the heat structures in the code input and determined that the appropriate heat structures were included. The response listed the heat structures and provided the information requested. The response to Question 15.06.05-32 is acceptable. In addition, the staff determined that the requirement of 10 CFR 50 Appendix K requirement I.A.6. “Reactor Internals Heat Transfer” was fulfilled. The heat transfer from piping, vessel walls, and non-fuel internal hardware was taken into account.

<u>Appendix K Compliance</u>

The applicant used conservative modeling techniques to assure that the results were bounding and comply with the requirements of 10 CFR 50 Appendix K and Three Mile Island (TMI) Action Items II.K.3.5, II.K.3.30 and II.K.3.31.

- Use of ANS-1971 x 1.2 fission product decay curve.
- Use of the Baker-Just correlation (not steam-limited) for metal-water reaction rate calculations.
- ZIRLO™ burst model (no burst was calculated to occur for any SBLOCA events).

- Moody model for choked-flow calculations of two-phase break discharge.

- Prevention of return to nucleate and transition boiling heat transfer modes for the initial blowdown phase.

- A top-skew axial power shape was chosen because it provided the distribution of power versus core height that maximized the PCT.

- The limiting single failure was assumed, which is the loss of one ECCS train, with one additional train out of service for maintenance. In this case, only two SI pumps are available.

- Minimum ECCS safeguards were assumed, which resulted in the minimum delivered ECCS flow available to the RCS.

- LOOP was assumed to occur simultaneously with the reactor trip, resulting in the delay of SI pumps and EFWS operations. RCP trip is assumed to occur 3 seconds after the reactor trip, as described in DCD Tier 2 Section 15.0.0.7.

- Shutdown reactivities resulting from fuel temperature and void were given their minimum possible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors expected during plant operations.

  Control rod insertion was considered to occur and assumed in the analysis with the most reactive single control rod postulated to be fully withdrawn.

In RAI 352-2369, Question 15.06.05-1, the staff requested the applicant to discuss all of the TMI Action Plan items listed in the SRP 15.6.5, and if a requirement was not applicable to state why. The applicant responded [Reference 15-7] by listing seven action items from the SRP and addressing each of the items. Item II.E.2.3 on Uncertainties in Performance Predictions was addressed by use of an ECCS evaluation model for the US-APWR during small break LOCAs that utilizes conservative Appendix K to 10 CFR 50 models. The staff review of MUAP-07013-P, “Small Break LOCA Methodology for US-APWR,” [Reference 15-11] confirmed that uncertainties in SBLOCA analysis were properly accounted for in determining the acceptability of US-APWR's ECCS performance pursuant to Appendix K of 10 CFR 50.

Item II.K.2.8, “Continued Upgrading of Auxiliary Feedwater (AFW) System,” and item II.K.3.40, "Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA," apply only to B&W designed reactors and therefore are not applicable to the US-APWR. Item II.K.3.5 "Automatic Trip of Reactor Coolant Pumps,” was also the subject of Question 15.06.05-16. In Question15.06.05-16, the staff asked the applicant if any of the Generic Letters 85-012, 86-005, and 86-006, “Implementation of TMI Action Item II.K.3.5, ”Automatic Trip of Reactor Coolant Pumps,” as noted in SRP 15.6.5, were applicable to the US-APWR design. If yes, the applicant was requested to provide an explanation for meeting the requirements for operation and tripping of reactor coolant pumps during SBLOCA. The applicant [Reference 15-7] responded that the guidance in Generic Letter 85-012, the Generic Letter applicable to Westinghouse plants, was followed and therefore the requirements of TMI action item II.K.3.5, ”Automatic RCP Trip during a LOCA,” were met. In the US-APWR, an automatic RCP trip will occur on an ECCS actuation signal generated from low pressurizer pressure, or high containment pressure. In the case of LOOP, the RCPs automatically trip after a three-second delay following the reactor trip. No

operator action is required to trip the RCPs during an SBLOCA. This explanation provided the information requested by the staff. The responses to Question 15.06.05-16, and also the portion of Question 15.06.05-1 pertaining to TMI Action Item II.K.3.5, are acceptable.

Regarding Item II.K.3.25, “Effect of Loss of AC Power on Pump Seals,” the applicant responded that for the US-APWR, RCP seals were designed such that the pressure tightness (or leak tightness) is usually maintained by the No.1 seal. In case of a failure of the No.1 seal, the No.2 seal can withstand full pressure as the defense-in-depth function. The applicant stated that RCP seal integrity is discussed in Chapter 8, “Electric Power,” Section 8.4.2.1.2 and Chapter 9, “Auxiliary Systems,” Section 9.2.2 of the DCD. **The ability of the No. 2 seal to withstand full pressure is still under staff review therefore Open Item 15.6.5-3 has been created in this SE to its track closure.**

TMI Action Items II.K.3.30, “Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K,” and II.K.3.31, “Plant-Specific Calculations to Show Compliance with 10 CFR 50.46,” are addressed by use of an M-RELAP5 based ECCS evaluation model that conforms to the requirements of 10 CFR 50, Appendix K, as established by the staff's acceptance in the MUAP-07013 SE [Reference 15-11] and by the SBLOCA analyses presented in DCD Section 15.6.5., “Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary.” The applicant addressed all seven TMI Action Items listed in SRP Section 15.6.5 and the response to Question 15.06.05-1 is acceptable.

In RAI 352-2369, Question 15.06.05-17, the staff asked the applicant what steps were included in the Emergency Operating Procedures (EOPs) to provide explicit guidance on safe restart of the RCP during a SBLOCA.

In the response [Reference 15-7], the applicant noted that restart of the RCPs was not explicitly addressed in the SBLOCA analyses, but will be incorporated into the Emergency Response Guidelines. The information provided by the applicant addressed the EOPs. The response to Question 15.06.05-17 is acceptable.

In RAI 352-2369, Question 15.06.05-18, the staff asked the applicant to explain the discussion of reactor coolant pump (RCP) trip in DCD Tier 2 Section 15.6.5.2.2, “Description of Small Break LOCA” which appeared to be inconsistent with the assumed LOOP concurrent with reactor trip. The discussion said RCP trip occurs three seconds after the ECCS actuation signal while the analysis showed RCP trip occurs three seconds after LOOP (reactor trip). In response to Question 15.6.5-18 [Reference 15-7], the applicant modified the wording in DCD Tier 2 Section 15.6.5.2.2 to clarify that the RCPs trip after a three second delay on the reactor trip signal eliminating the inconsistency noted by the staff. The response to Question 15.06.05-18 is acceptable.

RAI 352-2369, Question 15.06.05-25 requested the applicant to discuss how the grid spacers in the core were modeled, how the input data were determined (such as cross sectional areas and friction factors, etc.), and how the model and input data were validated. The M-RELAP5 modeling was validated using the Rig Of Safety Assessment/Large Scale Test Facility (ROSA/LSTF) SBLOCA, Oak Ridge National Laboratory/Thermal Hydraulic Test Facility (ORNL/THTF) uncovery heat transfer and flooding test data. The ORNL/THTF uncovery heat transfer test analysis demonstrated that M-RELAP5 tends to provide higher peak cladding temperature (lower heat transfer coefficient) than the measured data. The applicant’s discussion clarified how grid spacers were modeled and how input was determined. The

response to Question 15.06.05-25 is acceptable. The response to this RAI confirmed that the 10 CFR 50 Appendix K requirement I.C.2 “Frictional Pressure Drops” was satisfied.

In RAI 352-2369, Question 15.06.05-24, the staff requested the applicant provide an explanation of how the ECC water/steam interaction issue was handled In the M-RELAP5 model. In the response, the applicant stated [Reference 15-7] that the ECC coolant is provided from the DVI line to single phase liquid or two-phase mixture conditions, and therefore the strong water/steam interaction and condensation that is characteristic of PWRs with cold leg injection is not possible.

While the applicant's response did not directly answer the question, from the response the staff acknowledges that direct injection into the downcomer in effect eliminates the type of ECC water/steam interaction that can occur with cold leg injection. The staff agrees that injection into the downcomer nodes, where there is saturated liquid, significantly reduces the impacts of ECC water/steam interaction on the analysis results, and further notes that the nodalization used for the US-APWR plant model was the same as used for the integral test facility assessment cases, where the interaction of ECC water/steam was satisfactorily modeled based on the comparisons with the experimental data. Since the applicant cited confirmation from the code assessments that ECC water/steam interaction was satisfactorily modeled, and the staff concurs that this phenomenon does not significantly influence the plant response, the explanation provided by the applicant is acceptable. The response to Question 15.06.05-24 is acceptable. Nodalization studies performed by the applicant provided assurance that the 10 CFR 50 Appendix K requirement I.C.1.d, “Noding Near the Break and the ECCS Injection Points” was satisfied, i.e., the noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown. Further discussion on acceptability of the noding and time step size is in the SE [Reference 15-11] for the applicant's SBLOCA Methodology.

<u>Modeling Plant Specific Features</u>

Modeling techniques used to address specific US-APWR design features included:

- Empirical correlations to model the advanced accumulator characteristics.

- SI water temperature increases because the makeup water from the RWSP is recirculated. The temperature rise in the RWSP water is modeled.

The total uncertainty of the empirical correlations for the accumulator flow rate coefficient used for the safety analysis of the US-APWR was derived from instrument uncertainty, data scatter, manufacturing error, and scale bias. The uncertainty analysis is under review by the NRC as part of its review of the Advanced Accumulator Topical Report, MUAP-07001 [Reference 15-8].

The completion of the accumulator uncertainty analysis review does not impact the review of the advanced accumulator model in M-RELAP5; however, these uncertainties need to be considered in the US-APWR SBLOCA safety analysis to determine the advanced accumulator flow rate. **An Open Item 15.6.5-4 has been created in this SE to verify that the accumulator flow rate bias used in the SBLOCA evaluation is conservative relative to that determined in the Advanced Accumulator Topical Report, MUAP-07001 [Reference 15-8], which is still under staff review**.

Solution Convergence

In RAI 352-2369, Question 15.06.05-33, the staff questioned why the maximum time step sizes used for the 7.5- inch and 1-ft^2 break analyses were larger than the time step size for the DVI line break analysis. The staff expected larger break sizes would require smaller time steps due to more rapid changes. The applicant responded [Reference 15-7 and 15-28] that the time-step size required to simulate each transient differed because the phenomenon of interest and the location providing the Courant time-step limitation can be different. The applicant determined the maximum time step sizes used for the break cases based on sensitivity analyses. In DCD Tier 2, Table RAI 15.6.5-33.1, "Maximum Allowable Time-Step Sizes for US-APWR SBLOCA Sensitivity Calculations," the applicant showed the relation between the break position, location and the maximum allowable time-step determined for the US-APWR SBLOCA sensitivity calculations described in MUAP-07025-P (R2). The applicant's response provided the explanation requested. The response to Question 15.06.05-33 is acceptable. Acceptability of the noding for the M-RELAP5 SBLOCA model is documented in Reference 15-11.

The staff noted, in RAI 352-2369, Question 15.06.05-34, that in MUAP-07025-P, Section 5, "Analysis Results" most of the plots were long-term (1,000-5,000 seconds). However, many of the rapid changes occurred during the initial short term periods of time and they were difficult to determine in the long-term plots.

In response to this question, the applicant provided [Reference 15-7] short term plots for the pressure, power, flow and PCT during the initial 500 seconds. These plots for the rapidly changing variables allowed the staff to identify the rapid changes that occur early in the events, and provided the information requested. The applicant's response to RAI 15.06.05-34 is therefore acceptable.

Analysis Results

One consequence of stratification in a large horizontal pipe is that the properties of the fluid flowing through a small flow path in the pipe wall (i.e., a small break), called an off-take, depends on the location of the stratified liquid level in the large pipe relative to the location of the flow path in the pipe wall. If the off-take is located at the bottom of the horizontal pipe, liquid will flow through the off-take until the liquid level starts to approach (but not reach) the bottom of the pipe, at which time some vapor/gas will be pulled through the liquid layer and the fluid quality in the off-take will increase. If the phase separation phenomenon is ignored, vapor/gas will be passed through the off-take regardless of the liquid level in the pipe.

Likewise, if the off-take is located at the top of the pipe, vapor/gas will be flowing through the off-take until the liquid level rises high enough so that liquid can be entrained from the stratified surface. The flow quality in the off-take will decrease as the liquid level rises. If the phase separation phenomenon is ignored, liquid will pass through the off-take for all stratified liquid levels regardless of their height relative to the off-take. Lastly, if the off-take is located in the side of the large horizontal pipe, the same phenomenon of vapor/gas pull-through or liquid entrainment will occur, depending on the elevation of the stratified liquid level in the pipe relative to the location of the off-take in the wall of the pipe.

The RELAP5-3D stratification entrainment/pull-through model, for horizontal volumes, accounts for the phase separation phenomena and computes the mass and energy flowing through the off-take attached to a horizontal pipe when stratified conditions occur in the horizontal pipe.

This model is sometimes referred to as the off-take model. This model is used in M-RELAP5 to address the break enthalpy requirement in Appendix K Section I.C.1b for SBLOCAs.

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

In DCD Tier 2 Section 15.6.5, Revision 1 [Reference 15-1], the applicant reported on the results for three limiting small break LOCA cases:

- 7.5-inch upside break, the limiting break for PCT during the loop-seal clearance phase.
- 1-ft^2 upside break, the limiting break for PCT during the boil-off phase.
- 3.4-inch DVI line break, with only 1 train of SI system assumed to operate.

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

To confirm the M-RELAP5 results obtained by the applicant, the staff performed a series of audit calculations for the US-APWR SBLOCA [Reference 15-20 and 15-31] using the RELAP5/MOD3.3 computer code [Reference 15-30].

RELAP5/MOD3.3 is an advanced thermal/hydraulic simulation tool developed by the staff. Conservative assumptions were used in the RELAP5/MOD3.3 analyses similar to those used in the M-RELAP5 analyses. Decay heat was set at 120 percent of the ANS 1971 Standard. The single failure of one of the ECC trains was assumed.

In SBLOCA RAI Question CA-5 [Reference 15-18], the staff noted that confirmatory runs with RELAP5/MOD3.3 showed a large difference in PCT (approximately 300 °F (166.7 °C) lower to 200 °F (111.1 °C) higher) depending on the geometry of the stub pipe (length and area). In response to Question CA-5 [Reference 15-18] and follow-up meetings [Reference 15-25], the applicant made two revisions to the M-RELAP5 code. Version 1.5 included modifications that allowed use of the offtake model and the critical flow model at the same junction, therefore eliminating the need for the stub pipe. While use of the stub pipe was no longer necessary, the applicant retained the stub pipe in the calculations performed with M-RELAP5 Version 1.5. The staff performed additional confirmatory calculations with the RELAP5/MOD3.3 computer code [Reference 15-30]. These calculations again showed significantly different PCT values compared to the M-RELAP5 results.

In the process of investigating the reason for these differences, the staff obtained the M-RELAP5 source code and performed calculations with modified versions of that code and the RELAP5/MOD3.3 code. It was determined that the critical flow switching logic in the M-RELAP5 code was such that the required Moody critical flow model was not being used at all times when the break flow was two-phase. Rather, the code was switching between the Henry-Fauske and Moody models. Therefore, the 10 CFR 50 Appendix K requirement to use the Moody critical flow model whenever the conditions at the break are two-phase was not being met. The applicant revised the switching logic and corrected several other minor code problems in a new version, M-RELAP5 Version 1.6. This version of M-RELAP5 was used to produce the results in DCD Tier 2 Revision 3 [Reference 15-46].

The sensitivity cases in MUAP-07025-P [Reference 15-47] and the affected assessment cases in MUAP-07013-P [Reference 15-9] were also rerun with M-RELAP5 Version 1.6 to produce Revision 2 of MUAP-07025-P and Revision 2 of MUAP-07013-P. The applicant continued to use the stub pipe in the US-APWR plant calculations. However, with the critical flow switching logic corrected, the variation of PCT with stub pipe geometry was significantly reduced from 250 °F (139 °C) to 38 °F (21 °C). Also, when run with Version 1.6, the bottom of cold leg break became the limiting case rather than the top of cold leg break case, making the US-APWR results consistent with those of other PWRs where the bottom break is limiting. The calculations provided by the applicant for SBLOCA response in Revision 3 of DCD Tier 2 Section 15.6.5 [Reference 15-46] are now consistent with the confirmatory calculations and are acceptable.

In comparison with other four-loop PWR designs, the break size associated with the most limiting SBLOCA case for the US-APWR (i.e., 13.5 inches) is significantly larger. Typically the SBLOCA limiting break diameter for conventional four-loop PWRs is on the order of 2 to 4 inches. It was important to understand the reason that the limiting break size was larger for the US-APWR. The staff issued SBLOCA RAI question CA-1 [Reference 15-18] to address this issue. In the response, [Reference 15-25], the applicant listed and analyzed the effects of three unique US-APWR design features with the potential to significantly affect PCT limiting break size:

(1) an enhanced high-head safety injection (HHSI) system,

(2) a longer delay time for availability of the emergency power source for the safety injection system, and

(3) larger SGs (tube heat transfer area, tube flow area, etc.).

The analyses were based on power-scaled and geometry-scaled differences between US-APWR and conventional PWRs. The applicant's analysis demonstrated that fuel heatup was reduced, or did not occur, for break diameters smaller than 7.5 inches because of the much higher HHSI flow in the US-APWR relative to conventional PWRs. The applicant's analysis showed only small effects from the longer emergency power source delay and the larger total SG tube flow area in US-APWR. An independent sensitivity evaluation [Reference 15-20], using RELAP5/MOD3.3 [Reference 15-30] for the effect of enhanced HHSI flow in US-APWR was presented in Section 3.6 of the SBLOCA confirmatory analysis reports [References 15-20 and 15-31]. The findings of the RELAP5/MOD3.3 evaluation confirmed the findings presented in the applicant's analysis. When the ECCS flow was scaled back to the same flow to power ratio as existing PWRs, fuel heatup occurred in the break size range of two to four inches. The applicant's response to question CA-1 identified the design differences that make the US-APWR SBLOCA response different from existing PWRs and is acceptable.

Several aspects of the applicant's US-APWR SBLOCA model were different than typically used in PWR plant system models. For example, a "three-by-one" lumped loop model, combining the three intact coolant loops (i.e., those not containing the break) into a single loop was used in the M-RELAP5 US-APWR plant model. Combining the intact loops into a single loop was a method often used in the past to reduce the number of nodes in a system model and decrease computer execution time. The issue with models of this type is that asymmetries among the intact loops that might occur during an SBLOCA cannot be observed. An example is loop seal clearing, which is difficult to predict. It is possible that only one or two of the three intact loop seals might be predicted to clear during an SBLOCA. With the combined-loop model, however, it is not possible to capture this behavior as the equivalent of three loop seals would be

predicted to clear if any intact loop seal clearing was predicted. The staff issued SBLOCA RAI question CA-2 to address this issue [Reference 15-18]. The applicant's response [Reference 15-24] described the results from more detailed analyses that included M-RELAP5 sensitivity calculations performed using a US-APWR plant model with all four coolant loops independently modeled. The applicant's analyses showed that the lumped loop nodalization resulted in the conservative prediction of the PCT for medium SBLOCA break sizes, during which variability was seen in the number of loop seals cleared when the PCT occurred during the loop-seal clearing phase. For larger breaks, including the PCT limiting 1.0-ft^2 cold leg break, all loop seals cleared because of the rapid RCS depressurization. The applicant's analyses included assessment of M-RELAP5 loop seal clearing behavior using experimental data from the ROSA-IV/LSTF representing four different SBLOCA break sizes. These assessments showed M-RELAP5 was capable of predicting the loop seal clearing behavior in the experiments and conservatively predicted the PCT. The applicant's analyses also included assessment of M-RELAP5 capabilities for predicting the liquid remaining in the loop seals following clearing, using data from an experiment in the Upper Plenum Test Facility (UPTF). This assessment showed conservative M-RELAP5 capabilities, with more liquid predicted to remain in the loop seal (and therefore not being transported into the reactor vessel and core) than observed in the test. The applicant's response to question CA-2 is acceptable. It is noted that the RELAP5/MOD3.3 SBLOCA confirmatory calculations employed [Reference 15-20 and 15-31] a model that represented all four of the US-APWR reactor coolant loops.

An issue identified by the staff from not having a direct connection between the upper head and upper plenum was that the fluid temperature in the upper head region remained within a few degrees of the cold leg fluid temperature (551 °F, 288 °C) during steady-state initialization of the RELAP5/MOD3.3 SBLOCA model. The water temperature in the core upper plenum region was about 620 °F (327 °C).

It did not seem plausible that there would be a ~70 °F (~39 °C) temperature difference between the water in the upper head and upper plenum during normal full-power operation. With the M-RELAP5 modeling approach, the upper head temperature may be artificially low since the flow to the upper head from the upper part of the downcomer is influenced only by the temperature of the cold leg and is otherwise thermally isolated from the upper plenum region of the model. As a result, the temperature in the core region might be underpredicted during SBLOCA depressurizations, when the water in the upper head flashes and flows downward through the guide tubes. The staff issued SBLOCA RAI CA-4 [Reference 15-18] requesting further information on the expected temperature distribution in the upper regions of the US-APWR reactor vessel during steady full-power operation and a comparison with the M-RELAP5 predicted thermal conditions. In its response [Reference 15-24], the applicant indicated that during normal operation there is little upward flow through US-APWR control rod guide tubes and that the upper head water temperature is near the cold leg temperature. M-RELAP5 modeling for the upper reactor vessel region that provides an upper head temperature near the cold leg temperature was therefore justified. The applicant's response to question CA-4 provided the information requested and is therefore acceptable.

RAI 352-2369, Question 15.06.05-22 requested the applicant to resolve differences in the timing of events and PCT in the DCD Tier 2, Revision 1 and the sensitivity studies reported in MUAP 07025-P, Revision 0. In response to this question the applicant noted [Reference 15-7] both of these documents have been revised since this RAI was issued. The staff verified that the reported timing of events and PCT was consistent between DCD Tier 2 Section 15.6.5, Revision 2 and MUAP 07025-P, Revision 2. The response to Question 15.06.05-22 is acceptable because the applicant corrected the documentation.

In RAI 352-2369, Question 15.06.05-35, the staff noted that it appeared the pressure decreased immediately on a break, even for a small break and asked whether the pressurizer heaters were included in the model. The applicant was requested to discuss the control logic of the pressurizer heaters, and their impact on the pressure immediately after the break and before the scram. The applicant stated that the M-RELAP5 model did not include the backup and proportional heaters. The pressurizer pressure control system attempts to maintain the rated pressure even during transients. However, the capacity of the pressurizer heaters is limited and insufficient to mitigate the initial rapid depressurization occurring in SBLOCAs [Reference 15-7]. The response to Question 15.06.05-35 explained the rapid pressure decrease and is acceptable.

In RAI 352-2369, Question 15.06.05-36, the staff noted that it appeared the power started to decrease before the scram and asked the applicant to explain the cause of this decrease. The applicant explained [Reference 15-7] that the initial power decrease was caused by the negative reactivity feedback due to the coolant moderator density decrease that resulted from the initial depressurization. The response to Question 15.06.05-36 explained the power decrease and is acceptable.

In response to RAI 352-2369, Question 15.06.05-37, the applicant explained the reason for the rapid oscillations in the collapsed level in the core and upper plenum during the period from about 600 to 1200 seconds for the 2-inch break. These oscillations were more prominent in the 2-inch break as compared to other size breaks. The applicant responded [Reference 15-7] that the oscillations were density-wave oscillations during two-phase natural circulation. Two figures, “Comparison of transients of the pressure at core inlet (A), the core full-height differential pressure P2 (B), the differential pressure from top of core to top of U-tube PI (C), the liquid mass flow rate at top of U-tube (D), the liquid mass flow rate at hot-channel inlet (E) and the
void fraction at hot-channel outlet and at top of U-tube (F)” were provided to explain the mechanism for the oscillations. The response to Question 15.06.05-37 explained the mechanism for the oscillations and is acceptable.

In RAI 352-2369, Question 15.06.05-38, the staff noted that the core upper region uncovery occurred much earlier for 1-ft^2 crossover-leg break, when compared to 1-ft^2 cold-leg break. This implied the crossover-leg break was more severe, which contradicts the applicant’s conclusion that the cold-leg break was limiting. In response [Reference 15-7], the applicant stated that in comparisons between the cold-leg and crossover-leg breaks, the hydraulic resistance of the reactor coolant pumps (RCPs) played an important role in determining the plant responses for each case. In the early phase of the crossover-leg break, the initial core uncovery started earlier in the upper portion of the core because less coolant from the upper plenum and upper head entered the core. Therefore, the injected coolant contributed to refilling the core for a longer period of time when compared to the cold-leg break case. After the RCS pressure decreased, the accumulator started to provide ECC coolant to refill the RCS. In the cold-leg break, a larger amount of the injected coolant was swept out through the break when compared to the crossover-leg break. In the crossover-leg break, the RCPs resisted the injected coolant from being swept out in the crossover-leg break case, and the PCT was lower than in the cold-leg break case. The response to Question 15.06.05-38 described the reasons for the differences in the response between the two break locations and is acceptable.

The staff noted that for the limiting case the mass flow rate from one SI pump was 170 lb/sec (77 kg/s). Therefore, one SI pump injected about 51,000 lb (23,100 kg) more water in 300

seconds into the system than for the two-pump case. However, the mass inventory figures for the sensitivity calculations (MUAP-07025-P, Figure 5.6.1-7, "RCS Mass Inventory for 7.5-inch Break (Bottom)" and 5.6.1-20 "RCS Mass Inventory for 1-ft^2 Break (Bottom)" did not show the expected trend of this much additional liquid inventory when two pumps were assumed to operate. In RAI 352-2369, Question 15.06.05-39, the staff requested the applicant discuss what happens to the additional injected water when two SI pumps operate. In response to Question 15.06.05-39 [Reference 15-7], the applicant explained that in the case with a single SI pump injection failure, the combined effects of reduced break flow and increased accumulator injection caused by the lower pressure mitigated the coolant inventory decrease. That is, there was additional water added in the single failure case from the greater accumulator flow and less inventory loss from the break flow. The response to Question 15.06.05-39 explained the inventory trend and is acceptable.

In RAI 352-2369, Question 15.06.05-40, the staff asked how the SI pumps were represented in M-RELAP5 by a pump model or by a time dependent junction/volume. [
(Proprietary information withheld under 10 CFR 2.390)
]. The response to Question 15.06.05-40 described the SI model used for the US-APWR and is acceptable.

RAI 352-2369, Question 15.06.05-41, requested the applicant explain why the accumulator flow showed rapid oscillations in MUAP-07025-P [Reference 15-12], Figure 5.1.3-4, "Accumulator and Safety Injection Mass Flowrates for Pressurizer Steam Phase Break" for the pressurizer steam phase break case. The applicant explained [Reference 15-7] that for the case of the pressurizer steam phase break, the accumulator flow oscillation resulted from steam condensation in the downcomer around the height of DVI injection nozzle. The applicant compared the pressurizer break and the cold leg break cases to show that the condensation in the downcomer, around the height of DVI injection nozzle, drives the accumulator flow rate oscillation and that the pressure oscillation was more likely to occur when the steam flow rate in this region is low. The applicant also noted that this oscillatory behavior never challenged the safety criteria. The applicant provided an explanation of the oscillations as requested. The response to Question 15.06.05-41 is acceptable.

The staff noted that the PCT for the 7-inch cold leg bottom break was higher than the initial temperature while for the other break sizes between 1-inch and 11-inch, the PCTs were all below the initial temperature (Table 5.2-1, "Spectrum of Peak Cladding Temperatures for Cold-leg Break" of Reference 15-12). In RAI 352-2369, Question 15.06.05-42, the staff requested the applicant explain the reason for the heatup of the cladding for the 7-inch break. The applicant's response to Question 15.06.05-42 [Reference 15-7 referenced Table RAI 15.06.05-42.1] which was not included in the response. However, after this RAI was issued and responded to, the break spectrum was rerun with the newer M-RELAP5 code Version 1.6. The results for the spectrum of PCTs for the cold-leg breaks are now in Table 5.2-1, "Spectrum of Peak Cladding Temperatures for Cold-leg Break" of MUAP 07025-P (R2) [Reference 15-14]. A heatup now results for break sizes of 6-inches through 8-inches. The applicant explained that this heatup during the loop seal clearing period was due to the depression of the core water level and fuel rod uncovery that occurred when liquid was held up in the SG U-tubes and SG inlet plenum. This resulted in a pressure differential, which depressed the core liquid level. When the loop seals cleared, this differential pressure was quickly reduced and the core was recovered. The core uncovery can be rapid and deep but is short in duration. Based on a review of the revised break spectrum calculations, it was apparent that the smaller break sizes did not exhibit a heatup because the RCS inventory remained high enough to prevent core uncovery due the smaller break flow. For the larger breaks, the loop seals clear quickly enough to prevent

heatup during this time period. For still larger breaks, a heatup can occur during the boiloff period. The applicant provided an explanation for the heatup mechanism as requested. The response to Question 15.06.05-42 is acceptable.

In RAI 352-2369, Question 15.06.05-43, the staff noted that MUAP-07013-P [Reference 15-33] presented only one test, the ROSAIV/LSTF Integral-Effects Tests (IETs) for four important phenomena identified in the PIRT, and there were only two tests for four other important phenomena. Only two phenomena were validated by three tests, two Separate-Effects Tests (SETs) and the ROSA-IV/LSTF IET. The applicant initially responded [Reference 15-7] that the number of tests used was limited, but the test data scalability was addressed in prior RAI responses from the applicant to MUAP-07013. Subsequently, the applicant added several additional integral tests, including two additional ROSA-IV/LSTF tests, and one LOFT and one Semiscale test to the assessments in MUAP-07013-P (R2) [Reference 15-32]. A FLECHT-SEASET reflood test was also added to the assessment matrix. Updated Table 4.4.2-1 of MUAP-07013-P(R2) now shows that each important phenomenon has been covered by at least five test cases. Question 15.06.05-43 was addressed by inclusion of these additional assessment cases. The response to Question 15.06.05-43 is acceptable.

The staff also asked in RAI 352-2369, Question 15.06.05-43 if sensitivity studies were performed for the code simulations of the assessment cases with regard to nodalization and time step size. The applicant responded [Reference 15-7] by noting that sensitivity studies were completed as indicated in responses to the following RAI questions on MUAP-07013-P (R0): Questions 8.1.4-5 for the UPTF Full-Scale SG Plenum CCFL Test and 8.1.5-7 for the Dukler Air-Water Flooding Test [Reference 15-34], and Question 8.1-5 for the ORNL/THTF and the ROSA-IV/LSTF [Reference 15-48]. Nodding and time step sensitivities were performed. The response to Question 15.06.05-43 is acceptable.

In RAI 352-2369, Question 15.06.05-55, the staff asked if M-RELAP5 included a boric acid concentration calculation option, and if it did, if this option was used in the SBLOCA analysis. In response to Question 15.06.05-55 the applicant indicated [Reference 15-7] that M-RELAP5 has the capability to model boric acid concentration. However, this option was not used in the US-APWR SBLOCA analyses. This response addressed the boron concentration option. The response to Question 15.06.05-55 is acceptable.

In RAI 513-4170, Question 15.6.5-57, the staff noted that there was a discrepancy between the description of Section 3.3.5 and Figure 3-5, “Nodalization of the ECCS Injection,” in MUAP-07025-P. The applicant responded that the description in MUAP-07025-P [Reference 15-12] Section 3.3.5 contained a typographical error. The staff verified that MHI corrected this typographical error in Revision 2 of MUAP-07025-P [Reference 15-14]. The response to Question 15.06.05-57 is acceptable.

In RAI 514-4040, Question 15.6.5-58 the staff noted that Section 2.2 of MUAP-07025-P [Reference 15-12] categorized Appendix K Requirement # 4, “Initial Stored Energy in Fuel,” as Category 2 (additional validation needed to be performed to be able to use the model). However, Section 7.1.2 of MUAP-07013-P [Reference 15-33] stated that an annular pellet-to-clad gap heat transfer model derived from the FINE fuel rod design computer code was implemented in M-RELAP5. The staff also noted that MUAP-07013-P categorized Appendix K Requirement # 4 as Category 1 (required models were missing and needed to be added to M-RELAP5). The applicant responded [Reference 15-22] by noting that Requirement #4 is Category 1 and Category 3 (appropriate inputs or sensitivity studies were needed) in MUAP-07013-P. Also this requirement belongs to Category 2 (appropriate inputs including noding

address the requirements) and Category 1 (models in M-RELAP5 address the requirements) in MUAP-07025-P. The staff verified that the documentation classified Appendix K Requirement #4 as noted in the response. The response to Question 15.06.05-58 is acceptable because the applicant corrected the documentation.

The definition of the Categories was different between MUAP-07013-P and MUAP-07025-P. The applicant corrected MUAP-07013-P (R2) [Reference 15-32] and MUAP-07025-P (R2) [Reference 15-14] as noted above. The staff verified that this was corrected in MUAP-07013-P (R2) and MUAP-07025-P (R3) [Reference 15-47] resolving the discrepancy noted in the RAI.

In RAI 514-4040, Question 15.6.5-59, the staff noted that Appendix K Requirement # 27, "Reflood Rate," was categorized in MUAP-07025-P [Reference 15-12] as Category 1. In MUAP-07013-P [Reference 15-33], however, Appendix K Requirement # 27, "Reflood Rate," was categorized as being addressed with code inputs. The staff also asked in the RAI whether there were any new models implemented in M-RELAP5 pertaining to the reflood rate calculation. The applicant responded [Reference 15-22] by noting that no model change was made for the reflood rate calculation. In MUAP-07013-P, Requirement #27 should belong to Category 2 rather than Category 3 because a validation study was required for Requirement # 27 to conform to Appendix K. The staff verified that the documentation classified Appendix K Requirement #27 as noted in the response. The response to Question 15.06.05-59 is acceptable because the applicant corrected the documentation [Reference 15-32].

The staff noted that in Table 2-1, "Appendix K Requirements and Compliance of M-RELAP5" of MUAP-07025-P [Reference 15-12], Section 2.2 stated that Appendix K Requirement # 29, "Refill/Reflood Heat Transfer," was not applicable to SBLOCA. However, during a SBLOCA the water level in the core may drop below the top of the fuel assemblies during the loop seal clearance and boiloff phases. PIRT Phenomenon 11 ranks "Rewet Heat Transfer" as HIGH for the loop seal clearance, boil-off, and recovery phases of a SBLOCA. Therefore, the staff issued RAI 514-4040, Question 15.06.05.60, because this statement required further clarification.

In response to Question 15.06.05-60 [Reference 15-22], the applicant noted that the refill/reflood heat transfer was calculated using the post-CHF heat transfer model in M-RELAP5 rather than the reflood model in RELAP5-3D or the FLECHT heat transfer correlations when the reflood rates were 1-in/s or higher. A validation study was required for Requirement #29 to demonstrate conformance with Appendix K. The reflood heat transfer calculation was validated against the ORNL/THTF High-Pressure Reflood tests and the low pressure FLECHT-SEASET Forced Reflood tests.

Because the reflood rates generated from the accumulator and SI pumps are greater than 1 inch/s for the US-APWR SBLOCA analysis, the requirement for low flooding rates is not applicable to the US-APWR SBLOCA analysis. The reflood velocities for US-APWR SBLOCAs were given in MHI's response to SBLOCA RAI Question 7-16 [Reference 15-48] on MUAP-07013-P [Reference 15-33], and were shown to be 3-inch/sec or greater. Therefore, Requirement #29 should belong to Category 2 for the post-CHF heat transfer correlation in MUAP-07013-P and Category 1 in MUAP-07025-P. The staff verified that the documentation classified Appendix K Requirement #29 as noted in the response. The response to Question 15.06.05-60 is acceptable because the applicant corrected the documentation [References 15-32 and 15-14].

In RAI 514-4040, Question 15.6.5-61, the staff asked for an additional explanation regarding the adequacy of the model noding near the ECC injection points and the effects of steam leakage

from the RPV upper head region into the downcomer. The applicant responded [Reference 15-23] by noting that sensitivity calculations with a finer noding near the DVI injection points were performed. These calculations showed that the impact of the noding scheme on the condensation rate from the ECC water was not significant, and the resultant PCTs were similar to those of the DCD cases. For the steam leakage effects, the sensitivity study on the downcomer upper region noding showed that the resultant PCT was similar to that in the DCD case for the 7.5-inch break size case; the resultant PCT was lower than that in the DCD case for the 1-ft^2 break case.

Therefore, the staff concurs that the current noding scheme for the DCD calculations is appropriate for the safety analysis. The information provided by the applicant is acceptable. Therefore, the response to Question 15.06.05-61 is acceptable

In RAI 514-4040, Question 15.6.5-62, the staff requested the applicant evaluate the variability of PCT with CCFL model coefficients (both for the hot leg region and the SG U-tube inlet) and justify the values used in the SBLOCA evaluation model. The applicant responded [Reference 15-23] by noting that sensitivity calculations in terms of the CCFL at the SG inlet plenum and in the SG U-tubes were provided in the response to SBLOCA RAI Question CA-1 [Reference 15-25]. The staff reviewed the response to Question CA-1 and found that the sensitivity study established that the CCFL coefficients used by MHI were justified based on the low variability of PCT to the CCFL coefficients. Also, the staff notes that the applicant used the full scale UPTF experimental data to derive the coefficients used in the CCFL model; therefore, the basis for the CCFL model is acceptable. The information provided by the applicant is acceptable. The response to the Question 15.06.05-62 is acceptable.

MUAP-07025-P [Reference 15-12] Section 4.1.8 stated that the accumulator nominal water volume was 2,150 ft^3 (60.88 m^3),excluding the ineffective water volume, whereas DCD Tier 2 Section 6.3.2.2.2 and DCD Tier 2 Table 6.3-5 listed the accumulator water volume as 2,126 ft^3 (60.20 m^3), excluding the ineffective volume. In RAI 514-4040, Question 15.06.05-63, the staff asked the applicant to explain the apparent discrepancy in the accumulator water volume provided in MUAP-07025-P, Section 4.1.8, and DCD Tier 2, Section 6.3, and to provide the accumulator water volume utilized in the SBLOCA evaluation model. In the response [Reference 15-22] the applicant explained that the value of 2,126 ft^3 (60.20 m^3) was a minimum value whereas 2,150 ft^3 (60.88 m^3) was the nominal value. The accumulator water volume, without the ineffective water volume, ranges from 2,126 to 2,179 ft^3 (60.20 m^3 to 61.70 m^3) and the nominal (reference) value is actually 2,152 ft^3 (60.88 m^3). The applicant clarified that the actual nominal value of 2,152 ft^3 (60.88 m^3) was used for the US-APWR SBLOCA analyses. Use of the nominal value is acceptable because the accumulator does not empty for SBLOCAs. This is a limitation placed on the applicant's SBLOCA methodology by the staff in Reference [15-11]. The staff verified that DCD Tier 2 Table 15.6.5-2 “US-APWR Major Plant Parameter Inputs Used in the Appendix-K based Small Break LOCA Analysis” was updated from 2,150 ft^3 to 2,152 ft^3 in Revision 2 of the US-APWR DCD [Reference 15-2]. The discrepancy in the accumulator water volume has been corrected. The response to Question 15.06.05-63 is acceptable because the applicant corrected the documentation.

In RAI 514-4040, Question 15.06.05-64, the staff noted that Section 5.1.1 subsection (1), “Results of 2-inch cold-leg bottom orientation break,” in MUAP-07025-P [Reference 15-12] included a statement that after about 10 minutes following the 2-inch cold-leg break, the collapsed downcomer level abruptly dropped, as shown in Figure 5.1.1.a-6. “Downcomer Collapsed Level for 2-inch Break (Bottom).” The applicant was asked to explain this rapid change in the downcomer level. In the response to Question 15.06.05-64 [Reference 15-22],

the applicant stated that the decrease in the liquid level was caused by the inventory loss through the break and that prior to this time the break was being fed by liquid from higher elevations in the RCS. When the inventory was such that the vessel was filled to the downcomer level, then the rapid drop occurs. Later, the level was nearly constant because the flow from the SI pumps balanced the break flow. The explanation provided the information requested. The response to Question 15.06.05-64 is acceptable.

The staff noted in RAI 514-4040, Question 15.06.05-65, that, as shown in Figure 5.1.1.a-3, "Liquid and Vapor Discharges through the Break for 2-inch Break (Bottom)" of MUAP-07025-P [Reference 15-12], during periods of concurrent liquid and vapor discharge through the break, large oscillations in the calculated flow rates appeared. The applicant was requested to explain the oscillations in the discharge flow, including code modeling effects and/or physical phenomena. The applicant responded [Reference 15-23] that the oscillations in the discharge flow were caused by oscillations in calculated void fraction at the break cell of the cold leg, which ranged between about 0.5 and 1.0. A detailed explanation was provided for the oscillations in the discharge flow for the 2-inch break case. The oscillations were caused by the large variation in the void fraction at the break cell. Although the discharge flow oscillated, no significant variation in the core liquid level was observed and no heatup occurred. This was the only case in the break spectrum that showed two-phase flow at the break cell for a long period of time. The 1-inch break had primarily liquid conditions while the 3-inch break had mostly vapor flow. Heatup was not calculated for either the 1-inch or the 3-inch break. Therefore, the staff concluded that the calculated oscillations were not safety-significant, and that the information provided by the applicant in the response explained the oscillatory break flow rates predicted for the 2-inch break case and is acceptable. The response to Question 15.5.6-65 is acceptable.

In RAI 514-4040, Question 15.6.5-66, the staff requested an explanation for the relationship among the three collapsed liquid levels for: (1) the average fuel assembly; (2) the hot rod assembly; and (3) the upper plenum, shown in Figure 5.1.1.a-16, "Core and Upper Plenum Collapsed Levels for 1-ft^2 Break (Bottom)" of MUAP-07025-P[Reference 15-12] and an explanation for the statement in Section 5.1.1 that the "… figure also implies that a remarkable core uncovery occurs ..." It was noted that the collapsed level in the core regions can drop below the top of the core because liquid is held up by the upper core plate. In response to this question [Reference 15-22], the applicant noted that rapid depressurization and coolant loss due to the break caused flashing and voiding of coolant in the core and upper plenum regions, which decreased their collapsed liquid levels. With the continued coolant loss from the RCS, the collapsed liquid levels decreased further in the core and upper plenum regions and the liquid coolant was completely depleted in the upper plenum around 100 seconds after the break initiation. In addition, the applicant noted that the phrase "remarkable core uncovery" applied to the 1-ft^2 cold leg break case relative to the other breaks because a larger decrease in the core collapsed liquid level occurred. This indicated that the mixture level was also lower and that the most severe core uncovery occurred for the 1-ft^2 cold leg break case. The staff reviewed the applicant's explanation and concluded that it provides the information requested. The response to Question 15.06.05-66 is acceptable.

In RAI 514-4040, Question 15.6.5-67, the staff noted that there was a discrepancy between the Section 5.1.3 text and the associated Table 5.1.3-1, "Sequence of Events for Pressurizer Steam Phase Break," in MUAP-07025-P [Reference 15-12]. Section 5.1.3 summarized the results of the steam phase pressurizer break and stated that there was a "slight core uncovery of about 4-ft for the pressurizer steam phase break"; however, Table 5.1.3-1 for the pressurizer steam phase break stated that core uncovery did not occur. In response to this question

[Reference 15-22], the applicant noted that in MUAP-07025-P the explanation for Figure 5.1.3-7, “Core and Upper Plenum Collapsed Levels,” in Section 5.1.3 was incorrect. Core uncovery did not occur during the pressurizer steam phase break. The 4-ft uncovery referred to the decrease in collapsed liquid level in the core. However, the mixture level remained above the top of the core as illustrated by the collapsed level in the upper plenum as shown in Figure 5.1-3-7. Figure 5.1.3-8, “PCT at all elevations for hot rod in hot assembly,” showed that no heat-up occurred during the transient. The staff notes that the applicant modified the incorrect sentence in Section 5.1.3 of MUAP-07025-P (R2), removing the discrepancy. The response to Question 15.06.05-67 is acceptable because the applicant corrected the documentation.

In RAI 514-4040, Question 15.6.5-68, the staff noted that the availability of off-site power affects the RCP trip time and the ECC equipment response in a manner that could affect PCT. The staff requested justification for analyzing only the limiting loop-seal and boiloff PCT cold-leg SBLOCA cases with offsite power available.

In response to this question [Reference 15-22], the applicant noted that Figures 5.6.2-3, “Liquid Discharge through the Break for 7.5-inch Break (Top)” and 5.6.2-16, “Liquid Discharge through the Break for 1-ft_2 Break (Top)” in MUAP-07025-P [Reference 15-12] showed that the increase in discharged liquid from continuing RCP operation (off-site power available) was small, both for the 7.5-inch and 1-ft^2 cold leg breaks. For the 2-inch hot leg break, earlier start-up of the pumped SI was preferable to mitigate the accident consequence, when compared with the increase in discharged liquid due to longer RCP operation in the case assuming LOOP. In cases with larger break sizes where a faster depressurization was expected, the difference in RCP trip timing between the cases assuming LOOP and the cases without LOOP became smaller where a slower depressurization was expected. This mitigated the consequence of the accident due to early startup of the pumped SI as confirmed in Section 5.6.2 of MUAP-07025-P (R2) [Reference 15-14]. The discussion provided by MHI explained why the cases with LOOP were more limiting and why all of the cases did not need to analyzed to reach this conclusion. The response to Question 15.06.05-68 is acceptable because the applicant explained why LOOP cases are limiting and why there was no need to analyze all breaks with offsite power available.

In RAI 514-4040, Question 15.6.5-69, the staff noted that MUAP-07025-P [Reference 15-12] Section 5.4.1 referred to the loop-seal phenomena dominating PCT for the 1-ft^2 top cold-leg break. Based on the results shown in the accompanying figures, the PCT appeared to occur during the boiloff period, not during the loop seal period. In addition, the staff requested an explanation for the apparent discrepancy in the description and results of the PCT occurrence relative to the loop seal or boil-off phase of the transient. In response to Question 15.06.05-69 the applicant stated [Reference 15-22] that the statement in Section 5.4.1 Part (2), last paragraph, was incorrect and has been corrected in MUAP-07025-P (R2) [Reference 15-14]. The revised statement is: “The results show that the noding of the cold leg in the broken loop is adequate to predict the upstream conditions of the break flow when the PCT occurs during the boiloff phase for the 1-ft^2 top-side cold-leg break.” The applicant noted that loop seal clearing phenomena were not important for the 1-ft^2 break case because of the rapid depressurization. The response to Question 15.06.05-69 is acceptable because the applicant corrected the documentation and provided an explanation regarding the information requested by the staff.

In RAI 514-4040, Question 15.6.5-70, the staff noted that the loop noding sensitivity study presented in Section 5.4.2 of MUAP-07025-P [Reference 15-12] showed that loop seal clearance was predicted to occur sooner with the finer noding model, resulting in no heatup

(PCT). The staff requested a comparative description of the loop seal period for both the base case and the sensitivity case, including the times for loop seal clearance and expanded figures of applicable parameters around the time period of loop seal clearance. In response to this question [Reference 15-22], the applicant noted that in the finer noding case the cladding heats up later compared with the base case because the upper plenum emptied later, and the core heat was removed by the downflow from the upper plenum before it emptied. Also, in the finer noding sensitivity case, the broken loop seal cleared earlier than in the base case. Therefore, the turnaround in the PCT was earlier. The effect of nodalization on the loop seal behaviors was assessed using data from UPTF Test 5 (SET) and the ROSA-IV/LSTF SB-CL-18 (IET). The sensitivity calculation in MUAP-07025-P [Reference 15-13] showed that the nodalization used for the US-APWR SBLOCA calculations gives a conservative prediction for PCT when compared with the finer nodalization. In addition, the applicant's response to SBLOCA RAI Question CA-1 [Reference 15-25] demonstrated, by means of several sensitivity calculations, that cladding heatup was not significant during the loop seal phase. The staff concluded that the RAI response was satisfactory and that no additional noding studies were needed to establish PCT variability with loop seal noding. The response to Question 15.06.05-70 is therefore acceptable.

In RAI 514-4040, Question 15.6.5-71, the staff asked the applicant to justify the use of single sensitivity studies to evaluate variability with noding detail and time step size in MUAP-07025-P [Reference 15-12] and to confirm that the time step solutions were sufficiently converged. In response to this question [Reference 15-23], the applicant noted that the noding scheme for the US-APWR SBLOCA calculations were determined based on the code assessment analyses using experimental data obtained in several IET and SET test facilities. Since the noding scheme was similar between the code assessment and plant calculations, it can be concluded that the code assessment results validated the adequacy of the plant noding scheme. The sensitivity calculations for the code assessment and US-APWR SBLOCA analyses, along with the noding sensitivity calculations in MUAP-07025-P, showed that the current noding scheme is appropriate and tends to predict conservative consequences, PCTs, in the SBLOCA analyses. In the response to this RAI, the applicant also listed 13 additional noding sensitivity calculations that were performed and reported to the NRC to complement the code assessment analyses and the noding sensitivity analyses in MUAP-07013-P [Reference 15-33] and MUAP-07025-P [Reference 15-12]. These additional noding and sensitivity studies were provided to the NRC in response to various RAIs that were listed in the response to Question 15.06.05-71. The response to Question 15.06.05-71 is therefore acceptable and justified for the US-APWR nodalization.

With respect to sensitivity studies for the specified maximum time step size, the applicant ran both limiting cold leg breaks (7.5 inch and 1 ft^2) with half of nominal time step size so that three points were evaluated to determine the sensitivity. The results confirmed that the nominal maximum time step size provided solutions that were converged with 1 and 3 ºF (0.6 and 1.7 C) differences in the PCT. The additional sensitivity calculation results for the limiting SBLOCAs provided by the applicant demonstrated that the current time step size resulted in a converged solution. The response to Question 15.06.05-71 is therefore acceptable because the applicant demonstrated time step convergence.

The staff evaluated the noding and model sensitivity studies reported in the 13 RAI responses cited in the response to Question 15.06.05-71, MUAP-07013-P and MUAP-07025-P, together with the time step size sensitivity studies described above and concluded that the 10 CFR 50, Appendix K, Part II, "Required Documentation," Item 2 requirement that solution convergence be demonstrated by model noding and calculational time step studies is satisfied.

In RAI 514-4040, Question 15.6.5-72, the staff noted that Figure 5.5.a-5, "Accumulator Injection Mass Flowrate for 7.5-inch Break (Top)" of MUAP-07025-P [Reference 15-12] showed accumulator injection flow oscillations at points beyond 400 seconds into the transient for the 7.5-in cold leg top break case. The applicant explained [Reference 15-22] the mechanism for the oscillations was a feedback effect that was due to the accumulator injection flow increasing the cold leg pressure and therefore decreasing the flow from the accumulator. When the cold leg pressure decreased due to flow out of the break, there was an increased flow from the accumulator due to the larger pressure difference between the accumulator and the cold leg. The increased flow then led to more mass flow into the system, which raised the pressure. This process repeated itself and was the cause of the oscillations. While the calculated oscillations were time step size dependent, the oscillatory behavior appeared after the core was quenched and recovered, resulting in no sensitivity in the PCT. The response to Question 15.06.05-72 is acceptable because it explained the mechanism for the observed oscillations and showed that they did not affect the PCT. However, the staff notes that in MUAP-07025-P (R2) [Reference 15-14] time step size sensitivity calculations were performed for the 7.5-inch cold leg bottom break, in place of the 7.5-in cold leg top break case. At the time this RAI was responded to, the top of the cold leg break was limiting. Also, code changes made in M-RELAP5 Version 1.6, which resulted in use of only the Moody critical flow model during two-phase blowdown, eliminated the accumulator injection flow oscillations, rendering this RAI no longer relevant.

The staff noted in RAI 514-4040, Question 15.06.05-73 that Section 5.5 in MUAP-07025-P [Reference 15-12] provided a time step size sensitivity study for a 1-ft^2 top cold-leg SBLOCA. Figure 5.5.b-3, "Liquid Discharge through the Break for 1-ft^2 Break (Top)" showed that the time step sensitivity case did not calculate several of the liquid discharge rate peaks at points beyond 300 seconds into the transient. When calculations were run with M-RELAP5 Version 1.6 the peaks in the liquid discharge rate did not occur as shown in MUAP-07025-P (R2) [Reference 15-14] Figure 5.5.b-3, and therefore this RAI is no longer relevant.

In RAI 514-4040, Question 15.6.5-74, the staff noted that Figures 8.2.1-37, "Heater Rod Surface Temperature (Test Data)" and 8.2.1-38, "Heater Rod Surface Temperature (M-RELAP5, Base Case)" of MUAP-07013-P [Reference15-33] showed the measured and predicted rod surface temperatures, respectively. However, it was difficult to distinguish the temperatures given at several elevations. The staff requested the applicant provide figures which clearly showed the temperatures. In response to this question [Reference 15-22], the applicant noted that the response to the present question was provided in the applicant's response to SBLOCA RAI, Question 8.2.1-14 in UAP-HF-09492 [Reference 15-24]. The response to this question was evaluated by the staff in the SE for MUAP-07013-P (R2) [Reference 15-11] where the staff found that the revised Figures 8.2.1-37 and 8.2.1-38 in MUAP-07013-P (R2) clearly showed the temperatures for each elevation and were therefore acceptable. The response to Question 15.06.05-74 is therefore acceptable.

In RAI 514-4040, Question 15.6.5-75, the staff noted that MUAP-07025-P, Figure 5.5.b-5, "Accumulator Injection Mass Flowrate for 1-ft^2 Break (Top)" showed that the two time step sensitivity cases for the 1-ft^2 top cold leg break case resulted in lower accumulator injection rates between approximately 175 seconds and 275 seconds into the transient, affecting inventory levels, core uncovery, and PCT results. The staff requested the applicant to assess the variability of the results with time step size and justify the choice of the evaluation model maximum time step size. The applicant acknowledged in the response [Reference 15-22] that the accumulator flowrate between approximately 175 seconds and 275 seconds and the quenching timing were different.

However, the differences did not affect the PCT because the PCT occurred at 166 seconds, earlier than the occurrence of the differences between the two cases. However, the staff notes that in MUAP-07025-P (R2) time step size sensitivity calculations were performed for the 1-ft^2 cold leg bottom break, in place of the 1-ft^2 cold leg top break. For the 1-ft^2 cold leg bottom break, which is now the limiting case, the difference in the accumulator flowrate between the two cases was insignificant and Question 15.06.05-75 is no longer relevant.

In RAI 514-4040, Question 15.6.5-76, the staff requested the applicant to explain the apparent discrepancy in the time delay of the RCP trip between the SBLOCA analysis assumption and the US-APWR design description in DCD Tier 2. MUAP-07025-P [Reference 12] Tables 5.6.2-1, "Sequence of Events for 7.5-inch Break (Top)" and 5.6.2-3, "Sequence of Events for 1-ft^2-Break (Top)" showed the RCP Trip for the non-LOOP cases occurring exactly 18 seconds following ECCS actuation, not 15 seconds as described in DCD Tier 2 [Reference 15-1] Section 7.3.1.5.1. In response to the question [Reference 15-22], the applicant noted that the 18-second value used in the MUAP-07025-P was the analysis value for SBLOCAs, which is [
(Proprietary information withheld under 10 CFR 2.390)
]. This clarified the apparent discrepancy. The response to Question 15.06-05-76 is therefore acceptable.

The staff, in RAI 514-4040, Question 15.06.05-77, asked the applicant to explain the discrepancy in PCT values for the 1-ft^2 and 7.5-inch top break cases reported in the US-APWR DCD Tier 2 Section 15.6.5 [Reference 15-1] and the values reported in MUAP-07025-P [Reference 15-12].

In MUAP-07025-P (R3) [Reference 15-47] and US-APWR DCD Tier 2 Section 15.6.5, Revision 3, [Reference 15-46] the calculations have been performed with the newer M-RELAP5 version 1.6 code and do not show any discrepancies and Question 15.06.05-77 is no longer relevant.

MUAP-07025-P, Section 5.4.1 part (1) stated that the accumulator injection rates for the base case and sensitivity case for the 7.5 -inch cold-leg top break noding study were in perfect agreement. However, in RAI 514-4040, Question 15.06.05-78, the staff pointed out that the results were not identical. The applicant MHI06 [Reference 15-22] revised the statement to clarify that the calculation results of the base case and the sensitivity case were similar in terms of transient profile, magnitude and duration. The staff verified that this change was made in MUAP-07025-P (R2) [Reference 15-14]. The response to Question 15.06.05-78 is acceptable because the applicant revised the documentation to clarify the similarity in the sensitivity studies.

The staff concludes that the applicant's analyses for the spectrum of small piping breaks in the reactor pressure boundary are acceptable and meet the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50, and that the calculated performance of the emergency core cooling system following a postulated SBLOCA is acceptable.

15.6.5.4.3 Post-LOCA, Long-Term Cooling

15.6.5.4.3.1 Introduction

US-APWR DCD Tier 2 Revision 3 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary” [Reference 15-46], provides long-term cooling analyses in accordance with 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” which requires that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. In particular, US-APWR DCD Tier 2 Revision 3 Section 15.6.5.2.3, “Description of Post-LOCA Long Term Cooling” [Reference 15-46], identifies two considerations in the post-LOCA long-term cooling: (1) maintaining long-term decay heat removal and (2) the potential for boric acid precipitation.

15.6.5.4.3.2 Summary of Technical Information

US-APWR DCD Tier 2 Revision 3 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary” [Reference 15-46], describes the US-APWR ECCS as consisting of the accumulator system, the high-head injection system (HHIS) and the emergency letdown system. The accumulators initially inject at large flow rates and then are automatically reduced to lower flow rates as the water level in the accumulators drop below the level of the internal standpipe. The reduced flow from the accumulators, together with the DVI flow from the SI pumps is sufficient to maintain the downcomer level to provide flow to the core during the reflood phase. Furthermore, US-APWR DCD Tier 2 Revision 3 Section 15.6.5.2.3, “Description of Post-LOCA Long Term Cooling” [Reference 15-46], states that “After the quenching of the core at the end of reflood phase, continued operation of the ECCS supplies borated water from the RWSP to remove decay heat and to keep the core subcritical.” Borated water from the RWSP is initially injected through the DVI lines. If left uncontrolled, boric acid concentration in the core may increase due to boiling and reach the precipitation concentration. Boric acid precipitation in the core has the potential to affect the core cooling. To prevent the boric acid precipitation, the operator must switch over one operating DVI line to the hot leg injection line, allowing simultaneous DVI and hot leg injection. In the case of a hot leg break, almost all ECCS water injected through the DVI lines passes through the core and exits at the break. As a result, the boric acid concentration in the core does not increase. Even after the switchover, sufficient ECCS water passing through the core for decay heat removal is assured, and that simultaneously prevents any increase in boric acid concentration in the core. In the case of a cold leg break, the ECCS water delivered through the DVI lines may not be effective in flushing the core. As a result, the boric acid concentration in the core may increase. After the switchover, almost all ECCS water injected into the hot leg passes through the core. Therefore, the boric acid concentration in the core decreases. The main focus of the US-APWR DCD Tier 2 Section 15.6.5 post-LOCA, long-term cooling evaluation is to determine the switchover time from DVI to the simultaneous DVI and hot leg injection mode to prevent the boric acid precipitation, assuring long-term cooling is achieved.

As explained in US-APWR DCD Tier 2 Revision 3 Section 15.6.5.3.1.3, “Post-LOCA Long Term Cooling Evaluation Model,” an analysis method with an appropriate evaluation model was applied to control the boric acid precipitation and to assure long-term cooling after both small- and large-break LOCAs.

The calculation method is based on a two control volume model where the first volume is the core, lower plenum and upper plenum, and the second volume is the RWSP.

Open Item 15.6.5-5

RAI 861-6062, Question 15.6.5-99

> US-APWR DCD Tier 2 Revision 3 Section 15.6.5.3.1.3, “Post-LOCA Long Term Cooling Evaluation Model,” states that the post-LOCA, long-term cooling evaluation model is similar to the model described in several references, including the following: “Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, “Post LOCA Long Term Cooling Model” due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit, NRC letter dated November 23, 2005, D.S. Collins to G.C. Bischoff.” Provide an explanation of how the subject document relates to the US-APWR post-LOCA, long-term cooling evaluation model and how any non-conservatisms are treated.

Applying the post-LOCA, long-term cooling input parameters described in US-APWR DCD Tier 2 Revision 3 Section 15.6.5.3.2, “Input Parameters and Initial Conditions,” the evaluation model was used to compute the boric acid concentration as a function of time. The calculations were performed for both DVI mode and combined DVI and hot leg injection mode. The calculations were also performed assuming atmospheric pressure in the case of an LBLOCA and 120 psia for an SBLOCA. The post-LOCA, long-term cooling boron precipitation results were presented in US-APWR DCD Tier 2 Revision 3 Section 15.6.5.3.3.3, “Post-LOCA Long Term Cooling Evaluation Results.” The results for both large- and small-break LOCAs showed that the operators must switch from DVI to both DVI and hot leg injection mode at about four hours in order to avoid boron precipitation.

15.6.5.4.3.3 Regulatory Criteria

Relevant requirements of the regulations for this area of review, and the associated acceptance criteria, are given in Section 15.6.5 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Section 15.6.5 of NUREG-0800.

- 10 CFR 50.46, as it relates to paragraph (b)(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 17, as it relates to onsite and offsite electric power systems so that safety-related SSCs function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

- *GDC 27, as it relates to the combined reactivity control systems capability*. The reactivity control systems shall be designed to have a combined capability, in

conjunction with poison addition by the ECCS, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

- GDC 35, as it relates to demonstrating that the ECCS would provide abundant ECC to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts. The analyses should reflect that the ECCS has suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities available such that the safety functions could be accomplished assuming a single failure.

- 10 CFR 50.67, Accident source term, as it relates to mitigating the radiological consequences of an accident.

- The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

Additional guidance is provided in:

- RG 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003.

- Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," NRC, September 2004.

- Bulletin No. BL-93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers."

- NUREG-0933, Section 2, Issue 185, Control of Recriticality Following Small-Break LOCAs in PWRs.

- NUREG-0933, Section 3, Issue 191, Assessment of Debris Accumulation on PWR Sump Performance (Revision 2).

The acceptance criteria to meet the above requirements are:

1. Calculated maximum fuel element cladding temperature does not exceed 1,204 °C [2,200 °F].

2. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry are such that the core remains amenable for cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

6. An analysis of a spectrum of LOCAs has been performed to assure boric acid precipitation is precluded for all break sizes and locations.

15.6.5.4.3.4 Technical Evaluation

The staff performed the post-LOCA, long-term cooling technical evaluation of US-APWR DCD Tier 2 Revision 3 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary” [Reference 15-46], to confirm that: (1) the core remains cooled for the duration of the two-phase, long-term cooling (LTC) phase, (2) the boron concentration in the core keeps the core subcritical, and (3) boron precipitation will not obstruct core coolant flow.

15.6.5.4.3.4.1 Long Term Cooling Core Mixture Level

US-APWR DCD Tier 2 Revision 3 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary” [Reference 15-46], does not demonstrate explicitly that the reactor core remains covered by two-phase coolant mixture during the LTC phase following a LOCA. Possible reformation of the loop seals as a result of ECCS injection during the US-APWR LTC phase can result in suppression of the core two-phase mixture level. If the core level drops below the top of the active fuel, cladding heatup and oxidation can occur. To address the associated safety concern, the staff requested that the applicant provide the results of a thermal hydraulic analysis quantifying the two-phase mixture level in the US-APWR reactor during the LTC phase under the most limiting break size, break location, and ECCS performance conditions. The request was issued as RAI 706-5339, Question 15.6.5-79. The applicant responded [Reference 15-36] that the two-phase mixture level is not typically calculated in a standard post-LOCA, long-term core cooling evaluation.

It was assumed that the mixture level does not fall below the bottom of the hot leg before switchover to combined hot leg and vessel injection. The applicant response then discusses the transients with loop seals blocked/clear. No loops are calculated to become sealed during the post-LOCA short term phase due to the high core steaming rate and the high flow resistance. As the steam rate decreases, one loop may become sealed by ECC backflow at approximately 30 minutes. The loop flow resistance then increases with the blocked loop seal, and the core steaming rate continues to decline leading to an additional loop becoming sealed at 2 hours. Time histories of the reactor vessel differential pressure (defined as downcomer □P – core □P) and loop □P were presented as figures in the applicant response. These figures show that the reactor vessel differential pressure is always above the loop differential pressure (except during the very early portion of the transient).

This indicates that the core two-phase mixture level is higher than the bottom of the hot leg.

A sensitivity calculation was performed where the safety injection fluid temperature was decreased. The lower fluid temperature decreases the core void fraction, leading to a lower reactor vessel differential pressure. This lower fluid temperature also decreases the core steaming rate, decreasing the loop differential pressure. As a result, the loop differential pressure did not exceed the reactor vessel differential pressure and the two-phase mixture level was maintained above the hot leg bottom elevation up until the switchover to combined hot leg and direct vessel injection.

A sensitivity calculation was performed to simulate a different core axial power shape. In the base case, a uniform linear power shape was assumed. However, various axial power shapes during operation may occur aside from a uniform distribution. A sensitivity calculation was performed where the axial power shape was changed such that the core average void fraction was decreased to 80 percent of the base case. The increased liquid in the core decreases the reactor vessel differential pressure and it fell slightly below the loop differential pressure for a brief moment two hours into the transient, when the number of loops sealed increases from one to two. The applicant stated that the two-phase mixture level was maintained above the hot leg bottom elevation up to the hot-leg switch-over time (4 hours), while the calculation showed otherwise (even if only for a brief moment). It is not clear to the staff that decreasing the core average void fraction is conservative as stated in the applicant response.

The conclusion of the applicant response states that "the two-phase mixture level is maintained above the bottom of the hot leg elevation even in the case of loop seal plugging during the post-LOCA, long-term core cooling phase." However, the applicant previously assumed that the mixture level does not fall below the bottom of the hot leg before hot-leg switchover.

To address the above findings, the staff formulated a follow-up RAI question tracked below as an Open Item.

Open Item 15.6.5-6

RAI 861-6062, Question 15.6.5-95 (Follow-up to RAI 706-5359, Question 15.6.5-79):

> In the response to RAI 706-5339, Question 15.6.5-79, the applicant stated that the axial power shape affects the core average void fraction. The applicant then performed a sensitivity calculation where the core average void fraction was reduced to 80 percent of the base case value. Provide a basis for the 80 percent value.
>
> Using a uniform axial power shape and reducing the core average void fraction may not be conservative relative to using different axial power shapes. For example, a top peaked power shape may result in the same core average void fraction and result in the top of the core being exposed. Provide the results of a thermal-hydraulic analysis quantifying the two-phase mixture level in the US-APWR reactor during the long-term cooling phase assuming the most limiting break size, break location, and ECCS performance conditions. The analysis should include loop seal piping becoming plugged as well as the most limiting axial power shape.
>
> In the axial power sensitivity, the reactor vessel pressure differential falls to a value just below the loop pressure differential for a short time. This implies that the core mixture level is below the bottom of the hot leg elevation.

> At the same time, the conclusion is that the two-phase mixture level is always maintained above the bottom of the hot leg elevation. How can the conclusion statement be made when the condition that is observed does not meet the requirement?

In a separate RAI, the staff asked that the applicant provide the results for assessments that compare the predictions from the level swell model, used in the US-APWR two-phase mixture level assessment during the post-LOCA, long-term cooling phase, against low-pressure level swell test data of relevance for the plant analysis. This request was issued as RAI 706-5339, Question 15.6.5-80 [Reference 15-35]. The response to this question was provided by the applicant on March 29, 2011 [Reference 15-37]. In this response, it was stated that the Yeh correlation was used directly to calculate the void fraction which is then used to calculate the volume of liquid in the mixing volume. The applicant provided a figure, which compares the predicted and measured void fractions for test data, including some at low pressure (20 psia [0.14 MPa]). However, as it is not clear how many tests were actually run at low pressure, the staff issued a followup question to RAI 706-5339, Question 15.6.5-80, tracked below as an Open Item.

Open Item 15.6.5-7

RAI 861-6062, Question 15.6.5-96 (Follow-up to RAI 706-5339, Question 15.6.5-80):

> Further information is needed on the Yeh correlation. The response to RAI 706-5339, Question 15.6.5-80 includes Figure 1, which compares predicted versus measured void fractions including test data at low pressure (20 psia [0.14 MPa]). However, it is not clear how many tests were actually run at this low pressure. Provide a figure showing clearly the comparison against low pressure test data and include a table that lists the test flow conditions and measured void for each data point used in assessing the correlation at low pressure.

15.6.5.4.3.4.2 Boron Dilution and Return to Criticality Following a LOCA

This section presents the review of the US-APWR DCD Tier 2 Revision 3 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary” [Reference 15-46], relevant to Generic Safety Issue (GSI)-185, “Control of Recriticality Following Small-Break LOCAs in PWRs.” GSI-185 deals with safety concerns related to potential conditions during the course of a small-break LOCA when a slug of deborated coolant can accumulate in the cold leg piping and adjacent areas as a result of reflux condensation cooling in the SGs. Upon resumption of the natural circulation in the RCS primary loops or RCP restart, a deborated water volume can be transported into the reactor vessel and core potentially causing return to criticality and fuel damage.

As US-APWR DCD Tier 2 Revision 3 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary” [Reference 15-46], does not include consideration of recriticality consequences from possible boron dilution, the staff issued RAI 352-2369, Question 15.6.5-56 to address this safety concern. In addition, RAI 352-2369, Question 15.6.5-55, reviewed previously in the SBLOCA part of this report, also relates to this subject and is therefore considered here as well.

In RAI 352-2369, Question 15.6.5-55, the staff asked if M-RELAP5 includes an option for boric acid concentration calculation. The staff also asked if, provided that the option was available, it was activated in the SBLOCA analysis.

In its response to RAI 352-2369, Question 15.6.5-55 [Reference 15-7], the applicant explained that, while both M-RELAP5 and RELAP5-3D include a boron tracking capability, the function was not used in the US-APWR SBLOCA analyses. As discussed in the following, the applicant relied on separate assessments to analyze boron dilution during SBLOCAs. Therefore, the staff finds the response to RAI 352-2369, Question 15.6.5-55, acceptable and the relevant part of this RAI question resolved.

In RAI 352-2369, Question 15.6.5-56, the staff asked if the applicant considered the potential for accumulation of unborated water in the cold leg during reflux condensation and the impact of its subsequent transportation into the core after the restart of an RCP (the so called "Finish scenario"). The staff requested a description of the results of such an analysis and, if the analysis was not performed, an explanation as of why the scenario was not considered. In its response to RAI 352-2369, Question 15.6.5-56 [Reference 15-7], the applicant referred to sensitivity study results described in Topical Report MUAP-07025-P Revision 0, "Small Break LOCA Sensitivity Analyses for US-APWR" [Reference 15-12]. It was explained that below the 1-inch break size the break flow was too small to terminate the natural circulation after the break initiation and the potential reflux period was quite limited. Above the 6-inch break, the large break flow rate caused significant RCS depressurization relative to the SG secondary side pressure, which made reflux condensation in the SG of no significance. Accordingly, MHI explained that long-term reflux condensation appeared probable for cold-leg break LOCAs in the break range from 2-inch to 4-inch, for which deborated water would accumulate in the RCS primary loops, and stated that "significant amount of the deborated water may flow into the core when the natural circulation is reestablished or RCPs are restarted after the potential long-term reflux condensation." Referring to an evaluation by the applicant, the RAI response cited a value for the minimum core boron concentration that was required to maintain the reactor subcritical. This value was used as the criterion for assessing the available margin to recriticality following the restart of natural circulation and the associated transport of diluted condensate towards the core inlet. It was explained that this value was based on the assumptions described in the RAI response. It was also stated that the uncertainty associated with the core criticality was taken into account in the evaluation. In addressing the possibility of core recriticality following small break LOCAs, the minimum core entry boron concentration during the process of dilute slug propagation towards the core was used to determine if the reactor would remain subcritical. The response stated that the calculated minimum core entry boron concentration yielded a safety margin of approximately 300 ppm when compared to the minimum core boron concentration required to maintain the reactor subcritical under the assumed core conditions. Based on the review of the response to RAI 352-2369, Question 15.6.5-56, the staff identified specific items that required additional information from the applicant to resolve the safety concerns related to GSI-185. To address these outstanding items, the staff issued RAI 718-5402, Questions 15.6.5-83, 15.6.5-84, 15.6.5-85, and 15.6.5-86 and RAI 719-5352, Questions 15.6.5-90 and 15.6.5-91, which are discussed in the following.

The response to RAI Question 15.6.5-56 provided in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1" stated that a "significant amount of the de-borated water may flow into the core when the natural circulation is reestablished or RCPs are restarted after the potential long-term reflux condensation." On March 17, 2011, the staff issued RAI 718-5402, Question 15.6.5-83, asking the applicant to describe the dilution scenarios involving RCP restart that have been considered in the US-APWR evaluation of core recriticality associated with the inherent boron dilution mechanism occurring during small break LOCAs and discuss the core recriticality consequences under the identified limiting conditions including the coupled thermal-hydraulic system response and conditions.

The applicant was asked, in particular, to describe the conditions in the RCS and in the primary loops at the time of pump restart, RCP restart timing considerations, and loop transient flow characteristics following pump restart.

On May 13, 2011, the applicant responded to RAI 718-5402, Question 15.6.5-83 [Reference 15-40], referring to Generic Safety Issue (GSI)-185, “Control of Recriticality Following Small-Break LOCAs in PWRs” with regard to the identified concern of potential recriticality following a SBLOCA due to accumulation of deborated water in the RCP suction piping generated by steam condensation in the primary side of the SG tubes during reflux condensation. Four consecutive phases of the deboration scenario representing a boron dilution transient following a SBLOCA were identified and briefly described in the response for: (1) blowdown, (2) natural circulation, (3) reflux condensation, and (4) deborated water transient. The last two phases are of particular importance for the deboration scenario. It was explained that following the break initiation, the SG secondary side is isolated by closing the condenser steam dump valves. As a result, the SG secondary side pressure rises to the safety valve setpoint of 8.936 MPa [1,296 psia] and steam is released through the safety valves. The RCPs automatically trip 3 seconds after the reactor trip if no offsite power is available or 18 seconds after the ECCS actuation signal if offsite power is available. The primary system depressurization ends when the pressure falls to a value slightly above the saturation pressure of the SG secondary side corresponding to the safety valve setpoint. Following the interruption of natural circulation, the RCS reaches a quasi-steady state condition during the ensuing reflux condensation phase. At this point, as explained in the response to RAI 718-5402, Question 15.6.5-83 [Reference 15-40], operator action is undertaken to proceed to reactor shutdown. In this procedure, the SG secondary side is depressurized by opening the MSDVs or the MSRVs until the primary coolant temperature falls to 176.7 C [350 °F] where the RHR system is actuated to continue the cooldown process at a rate less than 100 °F/hour to prevent mechanical thermal shock. During the reflux condensation phase, which can take place during the depressurization period, the deborated condensate eventually accumulates and fills the RCP suction piping and then flows to the reactor vessel along with the borated safety injection water. Two mechanisms that can lead to a deboration transient were identified. One is by restart of RCPs and the second is by resumption of natural circulation due to the increase in ECCS injection flow as the RCS depressurizes during the reflux condensation phase. As slugs of the deborated water are transported into the reactor vessel and towards the core, clean (unborated) condensate mixes with the stagnant borated water in the vessel downcomer and in the lower plenum and if the coolant boron concentration flowing into the core is lower than the critical value, a recriticality will occur following the deborated water transient. The RAI response stated that only a narrow range of break sizes between approximately 3.81 cm [1.5 in] and 6.35 cm [2.5 in] in diameter are potentially susceptible to the accumulation of deborated water in the RCP suction piping and referred to the response to RAI 718-5402, Question 15.6.5-85, for a description in this regard. The referenced response to RAI 718-5402, Question 15.6.5-85, is reviewed elsewhere in this report. Smaller breaks do not lead to interruption of the natural circulation before SG cooldown begins. For larger breaks, either the RCS depressurizes below the SG secondary side pressure and deborated water does not accumulate in the loops or no resumption of the natural circulation takes place during the reflux condensation phase and thereafter. Considering specifically the scenario with RCP restart, it was explained that this RCP restart scenario is not expected to occur due to the following reason: For the range of break sizes associated with reflux condensation, the RCS liquid inventory is depleted enough so that the pressurizer empties during the reflux condensation and the pressurizer liquid level starts increasing as the natural circulation recovers.

Thus, RCP restart is not expected during the time of reflux condensation and the scenario with natural circulation resumption is more probable to cause a rapid insertion of the deborated water into the vessel prior to the scenario of RCP restart. Furthermore, the applicant explained that even if the RCP restart scenario is taken into account, it cannot be the limiting case for the recriticality evaluation since the emergency procedure does not recommend a concurrent restart of multiple RCPs (it is possible to remove the decay heat by RCS circulation with a single RCP restart). Regarding the evaluation of recriticality consequences, the RAI response referred to the responses to RAI 352-2369, Question 15.6.5-56, and RAI 719-5352, Question 15.6.5-90, for an explanation and stated that the slugs of deborated water were assumed to flow simultaneously from all four loops to the reactor vessel to conservatively reduce the evaluated boron concentration at the core inlet. The referenced responses to these RAI questions are reviewed elsewhere in this document. The staff finds the clarifying information provided in the response to RAI 718-5402, Question 15.6.5-83, acceptable. In addition, taking into consideration that, in specific instances, the RAI response refers to responses to RAI 718-5402, Question 15.6.5-85; RAI 352-2369, Question 15.6.5-56; and RAI 719-5352, Question 15.6.5-90; reviewed elsewhere in this document, the staff considers the response to RAI 718-5402, Question 15.6.5-83, acceptable and the RAI question resolved.

The response to Question 15.6.5-56 given in UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1” [Reference 15-7] stated that “significant amount of the de-borated water may flow into the core when the natural circulation is reestablished or RCPs are restarted after the potential long-term reflux condensation.” On March 17, 2011, the staff issued RAI 718-5402, Question 15.6.5-84 [Reference 15-38], asking the applicant to describe the dilution scenarios assuming resumption of RCS natural circulation that have been considered in the US-APWR evaluation of core recriticality associated with the inherent boron dilution mechanism occurring during small break LOCAs and discuss the results from the analysis of the core recriticality consequences under the identified limiting conditions including the coupled thermal-hydraulic system response and conditions. The applicant was also asked to identify and describe the conditions that are found to lead to the worst core recriticality consequences, in particular to discuss: (1) the RCS loop conditions preceding natural circulation resumption, (2) the process of natural circulation resumption in individual loops, (3) timing aspects of interruption and resumption of natural circulation, and (4) loop flow transient characteristics during natural recirculation resumption and to provide the technical basis in support of the identified and applied limiting boron dilution conditions. If test data were used as part of the technical basis, the applicant was asked to demonstrate their applicability, sufficiency, and scaling with regard to the US-APWR reactor design.

On May 13, 2011, the applicant responded to RAI 718-5402, Question 15.6.5-84 [Reference 15-40], referring to the response to RAI 718-5402, Question 15.6.5-83, for the description of the boron dilution transient evolution.

The applicant explained that part of the condensate generated in the SG U-tubes during reflux condensation accumulates in the primary loops and fills the RCP suction piping. For simulation details regarding the plant behavior, the RAI response referred to the response to RAI 718-5402, Question 15.6.5-85, which is described in the next paragraph. Regarding the evaluation of recriticality consequences, it was explained that the natural circulation resumption was assumed to occur at the end of the reflux condensation phase. At this point, the RCS coolant temperature decreases to the lowest temperature for Mode 3 (hot standby) operation (176.7 °C [350 °F]), thus requiring the highest boron concentration to maintain subcriticality and the duration of reflux condensation is long enough to fill the loops with deborated water.

It was explained that while the analysis described in the response to RAI 718-5402, Question 15.6.5-85, indicated that the natural circulation resumption does not occur in all loops simultaneously, the bounding analysis described in the responses to RAI 352-2369, Question 15.6.5-56, and RAI 718-5402, Question 15.6.5-85, assumed a simultaneous start of the deborated water transient in all four loops and the total amount of deborated water entering into the reactor vessel was considered for the recriticality evaluation in the bounding analysis. As under such conditions, where all the clean condensate, assumed as already existent and being stored in any of the four RCS loops, would be reaching the reactor vessel upon the restart of natural circulation, the staff agrees that such a system response would be conservative for the purpose of assessing the recriticality consequences from a boron dilution transient. Therefore, the staff considers the response to RAI 718-5402, Question 15.6.5-84, acceptable and the RAI question resolved.

In the response to RAI Question 15.6.5-56 in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1," it was stated that a prolonged reflux condensation phase appears probable for US-APWR cold leg breaks that range between 2 and 4 inches of equivalent break diameter. It is further claimed that breaks below 1 inch do not lead to natural circulation interruption and break sizes larger than 6 inches depressurize the primary reactor system to a degree that precludes the occurrence of significant reflux condensation. On March 17, 2011, the staff issued RAI 718-5402, Question 15.6.5-85, asking the applicant to identify the US-APWR small break cases analyzed in the US-APWR evaluation of the inherent boron dilution mechanism during small break LOCAs, the US-APWR small break LOCA model and computer codes used to perform supporting analyses and to substantiate the sufficiency of the analyzed cases. In addition, staff asked the applicant to provide assessment results relevant to the boron dilution transient and explain how the results from performed thermal-hydraulic analyses were applied in the US-APWR boron dilution recriticality evaluation.

On May 13, 2011, in its response to RAI 718-5402, Question 15.6.5-85 [Reference 15-40], the applicant identified the code used in the analysis, discussed its applicability, described the models and conditions applied, and presented analysis results along with conclusions. The M-RELAP5 code, applied in the US-APWR small-break LOCA analysis, documented in MUAP-07013-P (R2), "Small Break LOCA Methodology for US-APWR," October 2010, was also used to analyze the SBLOCA cases for the evaluation of inherent boron dilution. The code was reviewed by the staff separately as part of the safety evaluation of MUAP-07013-P. It was explained that M-RELAP5 has been widely assessed using test data obtained in various facilities with various break sizes. Concerning very small break sizes in particular that are of interest for boron dilution due to occurrence of reflux condensation, Test SB-CL-12 (0.5 percent cold leg break equivalent to US-APWR 2.5 inch cold leg break) performed at the ROSA/LSTF was identified. The ability of M-RELAP5 to model counter-current flow in the SG primary side, importance for reflux condensation, was assessed using separate-effects test data obtained in the UPTF as well as data from the Dukler air-water tests as described in MUAP-07013-P.

The M-RELAP5 capability to simulate natural circulation and SG heat transfer was assessed using SBLOCA test data obtained in the ROSA facility. It was also explained that RELAP5-3D, the base code for the development of M-RELAP5, was validated with regard to reflux condensation during SBLOCAs using test data obtained in the ROSA/LSTF facility. Describing the analysis model, it was explained that M-RELAP5 was applied to investigate the boron dilution event in the same manner as in the ECCS performance evaluation described in US-APWR DCD Tier 2 Section 15.6.5.3.1.2, "Small Break LOCA Evaluation Model," with the following changes.

The plant noding scheme was changed such that the model can appropriately simulate reflux condensation and ensuing natural circulation resumption occurring in the loops. In addition, logic to control the water level on the SG secondary side for an accurate long-term simulation as well as control logic to regulate MSDVs or MSRVs on the SG secondary side to simulate the RCS depressurization by operator action according to the US-APWR ERG were implemented. Presenting the analysis results, it was stated that the US-APWR SBLOCA M-RELAP5 simulations indicated that the reflux condensation was possible within a range of break sizes between approximately 3.81 cm [1.5 in] and 6.35 cm [2.5 in] in diameter. Table No. RAI 15.06.05-85.1, in the RAI response presented the predicted transient times at start of SG cooldown, termination of SG cooldown, interruption of natural circulation, and resumption of natural circulation for 2.54 cm, 3.81 cm, 5.08 cm, 6.35 cm, and 7.62 cm [1.0 in, 1.5 in, 2.0 in, 2.5 in, and 3.0 in] cold leg breaks. In addition, the thermal-hydraulic system response for the 5.08 cm [2.0 in] break was discussed in detail. The predicted primary and secondary pressures were presented in Figure RAI 15.06.05-85.1 and the coolant temperatures at the core inlet were depicted in Figure RAI 15.06.05-85.2. Figure RAI 15.06.05-85.3 showed the liquid flow rates at the top of the SG U-tubes for each loop to illustrate the interruption and resumption of natural circulation. The natural circulation was interrupted in all four loops at approximately the same time. Figure RAI 15.06.05-85.4 depicted the coolant flow rates at the cold leg nozzles indicating the early natural circulation resumption in the broken loop. It was explained that the deborated condensate from the SG exit and the borated coolant from the vessel were facilitated to flow toward the break resulting in the early natural circulation resumption in the broken loop. The predicted timing of resumption of natural circulation in each loop was given in Table RAI 15.06.05-85.2. The amount of condensate generated in the uphill-side and downhill-side of the SG U-tubes was shown in Figures RAI 15.06.05-85.5 and 6, respectively. It was stated that the amount of condensate generated in the SG U-tubes downhill-side was large enough to fill the RCP suction piping with deborated water. Figure RAI 15.06.05-85.8 showed the downcomer collapsed level indicating that during the interruption of natural circulation the level remained stable around the elevation of the DVI injection nozzle or the cold leg. Based on the analysis results presented in the response RAI 718-5402, Question 15.6.5-85, the staff finds that the range of break sizes analyzed as well as the applied model were appropriate for the thermal-hydraulic analysis of the inherent boron dilution transient. The staff also agrees that it is reasonable to expect that the natural circulation would not resume simultaneously in all primary coolant loops as indicated by the code predictions. As the RAI response provided information regarding the thermal-hydraulic system response that needs to be considered in the analysis of the inherent boron dilution transient recriticality consequences for the critical range of break sizes, the staff considers the response to RAI 718-5402, Question 15.6.5-85, acceptable and the RAI question resolved.

The inherent boron dilution process can take place during the event of a small break LOCA in the US-APWR. When the water level in the reactor pressure vessel drops below the hot leg inlet and only steam starts flowing to the SGs, the natural circulation in the RCS will cease and will switch to a reflux condenser cooling mode. It is the reflux condenser cooling mode during which boron-depleted condensate is generated within the primary RCS via heat extraction to the secondary side through the SG U-tubes. In this mode, a fraction of the condensate flows from the vertical SG U-tubes toward the pump loop seal and the remaining fraction of the condensate returns back to the upper plenum via the hot-leg. In the hot leg, the returning condensate and the steam form a counter-current flow pattern. The counter-current flow of condensate and steam in the horizontal section of the hot leg and in the connected bend and inclined piping is possible only under a certain range of flow rates, which are limited by the counter-current flow limitation phenomenon. The ratio of the US-APWR core thermal output to that of a current four-loop PWR is 1.30, whereas the ratio for the hot leg inner diameter is 1.07.

Experimental evidence from PWR hot leg test facilities, including recent geometrically scaled tests at the Transient Two Phase FLOW TOPFLOW test facility, reveals that steam-liquid interaction processes in the horizontal hot leg piping as well as in the elbow and inclined section of the hot leg are described by their own distinct governing characteristics.

On March 17, 2011, the staff issued RAI 718-5402, Question 15.6.5-86, asking the applicant to present the experimental data base that validates the US-APWR small break LOCA methodology for modeling reflux condenser cooling of importance for the boron dilution analysis given that the US-APWR hot leg was not sized up proportionately to the reactor thermal power increase when compared to current US PWRs. The applicant was also asked to describe the relevant scaling methodology along with the scaling results for the US-APWR design, including consideration of flow conditions and parameters of governing importance for reflux condenser cooling including counter-current flow limitation.

On May 17, 2011, the applicant responded to RAI 718-5402, Question 15.6.5-86 [Reference 15-40]. The applicant explained that the M-RELAP5 model of the counter-current flow limitation in the hot leg was validated for US-APWR SBLOCA applications using test data obtained in the UPTF and referred to Revision 2 of MUAP-07013-P, "Small Break LOCA Methodology for US-APWR," in this regard. The UPTF steam-water test was conducted under 0.3 MPa [43.5 psia] and 1.5 MPa [217.6 psia]. It was also explained that the applicability of the M-RELAP5 CCFL model and its validation results were addressed by the applicant in the responses to SBLOCA RAI Questions 8.1.4-3 and 8.1.4-11 regarding the M-RELAP5 SBLOCA topical report [Reference 15-33]. The responses to these RAI questions were reviewed by the staff as part of the safety evaluation of MUAP-07013-P. Referring to SBLOCA RAI Question 8.1.4-11 in particular, the applicant stated in the response to RAI 718-5402, Question 15.6.5-86, that the effect of pressure was investigated using steam-water test data obtained at 1.5 MPa [217.6 psia], 3.0 MPa [435 psia], and 5.0 MPa [725 psia] in the TOPFLOW facility and it was concluded that the current M-RELAP5 model remained conservative for evaluating the actual plant, even under the pressure range expected for the reflux condensation period as analyzed in the response to RAI 718-5402, Question 15.6.5-85. The staff finds the information provided in the response to RAI 718-5402, Question 15.6.5-86 acceptable. In addition, taking into account that SBLOCA RAI Questions 8.1.4-3, 8.1.4-11 and RAI 718-5402 Question 15.6.5-85 has been resolved, the staff considers RAI 718-5402, Question 15.6.5-86, resolved.

In the response to RAI 352-2369, Question 15.6.5-56 provided in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1," July 2009, [Reference 15-7], the issue of inherent boron dilution during small break LOCAs in the US-APWR was discussed. Referring to an evaluation by the applicant, the RAI response cites [(Proprietary information withheld under 10 CFR 2.390)] to maintain the reactor subcritical. This value is used as the criterion for assessing the available margin to recriticality following the restart of natural circulation and associated transport of diluted condensate towards the core inlet. It was explained that this value was based on the assumptions that the xenon concentration in the core was at its equilibrium, the most reactive control rod assembly was stuck out of the core, and the initial reactor coolant system boron concentration was 1,000 ppm. It was also stated that the uncertainty associated with the core criticality evaluation was taken into account.

On March 17, 2011, the staff issued RAI 719-5352, Question 15.6.5-90, asking the applicant to provide the reactor core conditions that have been assumed in the criticality calculation for determining the minimum core boron concentration required to maintain the reactor subcritical and to quantify any conservative margins included in the calculated minimum core boron

concentration such as available shutdown margin and additions to the criticality result for conservatism. In particular, the applicant was asked to specify the reactor core temperature, reactor coolant pressure, and core life cycle point in time.

On May 19, 2011, the applicant responded to RAI 719-5352, Question 15.6.5-90 [Reference 15-41], and provided results from criticality calculations in Table RAI 15.06.05-90.2. The table lists values for the recriticality criteria in terms of core boron concentrations calculated for the beginning of cycle (BOC) and for end of cycle (EOC) of the 24-month core using a three-dimensional core simulator. In this regard, the RAI response referred to Revision 0 of MUAP-07019-P, "Qualification of Nuclear Design Methodology using PARAGON/ANC," December 2007. The critical boron concentrations were calculated for hot standby in Mode 3 assuming conservatively that the control rod with the maximum reactivity worth was stuck out of the core and accounting for the code's calculational uncertainty of 100 ppm equivalent boron content. In addition, an equilibrium xenon condition for the core was conservatively assumed although xenon would build up by the time the slugs of deborated water entered the reactor vessel. Thus, the recriticality criterion [(Proprietary information withheld under 10 CFR 2.390)

]. Besides the recriticality criteria, the initial boron concentrations in the RCS were also provided in the table. The initial critical boron concentration for BOC was obtained at HFP (hot full power), ARO (all rods out), and equilibrium xenon state resulting in 1,086 ppm rounded down to 1,000 ppm for conservatism. As the initial HFP boron concentration approaches zero ppm at EOC, a value of 0 ppm was appropriately given in the table. The staff finds that the applicant provided sufficient information to describe the conditions under which the recriticality criteria were obtained. In addition, the sources of conservatisms, as introduced by making specific analytical assumptions, were identified and the associated margins of conservatisms in establishing the recriticality criteria were provided. As this RAI question addressed the criticality calculation for determining the minimum core boron concentrations, based on the review of the provided information in the RAI response in this regard, the staff considers the response to RAI 719-5352, Question 15.6.5-90, acceptable and the RAI question resolved. As this part of the response to RAI 719-5352, Question 15.6.5-90, refers to Revision 0 of MUAP-07019-P, "Qualification of Nuclear Design Methodology using PARAGON/ANC," December 2007, the proposed closure of this RAI question is conditional on the approval of MUAP-07019-P by the staff. Therefore, an Open Item will track the resolution of this condition. Information provided in this RAI response related to the mixing model and related assessment results for the mixed boron concentration at the core inlet is addressed as part of the review of RAI 719-5352, Question 15.6.5-91, described in the next paragraph.

Open Item 15.6.5-8

RAI 719-5352, Question 15.6.5-90:

> The response to RAI 719-5352, Question 15.6.5-90, refers to Revision 0 of MUAP-07019-P, "Qualification of Nuclear Design Methodology using PARAGON/ANC," December 2007.

Therefore, the closure of this RAI question is conditional on the NRC's approval of MUAP-07019-P Revision 0. Provide the status of NRC's approval of MUAP-07019-P Revision 0.

According to a core recriticality evaluation for small break LOCAs, as described in the response to RAI Question 15.6.5-56 provided in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1," July 2009, the minimum core entry boron concentration during the process of dilute slug propagation towards the core was used to determine if the reactor will remain subcritical. It was stated in this response that the minimum core entry boron concentration remained sufficiently above the critical boron concentration determined as required to maintain the reactor subcritical under certain assumed core conditions. In addition, it was explained that immediate mixing between the borated reactor coolant and the diluted slug was assumed to take place only in the lower plenum when determining the minimum core entry boron concentration. On March 17, 2011, the staff issued RAI 719-5352, Question 15.6.5-91, asking the applicant to provide a detailed description of the analytical mixing model used to calculate the minimum core entry boron concentration during the dilute slug propagation process and discuss the conservatisms of the obtained results and substantiate the appropriateness of the applied approach. In addition, the staff asked, as appropriate, that the applicant include the modeling equations as well as any computer programs used to perform the calculations and provide all assumptions used to develop the model and to perform the calculations. In particular, the applicant should describe the initial conditions and provide the input values for the model calculations.

On May 19, 2011, the applicant responded to RAI 719-5352, Question 15.6.5-91 [Reference 15-42]. The applicant explained that the mixing model applied in the US-APWR boron dilution analysis assumed simplistic mixing of the liquid volume contained within the lower plenum with the total volume of clean deborated water assumed as being accumulated in all four primary loops during the reflux condensation and prior to the resumption of natural circulation following an SBLOCA. Thus, the boron concentration of the mixed coolant was simply calculated as a volume-weighted average using the volume of stagnant borated water in the lower plenum and the volume of deborated water stored in the loops. The lower plenum volume was determined from the RPV design. The total volume of deborated water was determined assuming that the clean condensate occupied the primary main coolant piping from the SG exit to the RCP suction inlet of all four loops. Finally, to calculate the volume-weighted average concentration, the applicant introduced assumptions with regard to the initial boron concentrations in both volumes prior to mixing. Thus, it was assumed that the deborated condensate in the loops was completely free of boric acid (0 ppm boric acid concentration). The staff agrees that this is conservative, as such an assumption would maximize the degree of deboration due to mixing. Further, as explained in Table RAI 15.06.05-90.1 in the response to RAI 719-5352, Question 15.6.5-90, the initial boron concentration of the stagnant liquid of the lower plenum was once again determined as a volume-weighted average value using the entire coolant volume of the RCS with no SG tube plugging and the minimum RWSP liquid volume of 2,208.7 m^3 [78,000 ft^3] at the minimum boron concentration of 4,000 ppm. The RCS coolant volume was determined from the RCS design. For the limiting BOC case, the RCS boron concentration was appropriately assumed at the critical boron concentration at HFP, ARO, and equilibrium xenon state of 1,000 ppm as provided in the response to RAI 719-5352, Question 15.6.5-90 (see Table RAI 15.06.05-90.2). The staff found that DCD Table 6.2.1-3, "RWSP Design Features," listed a value of 2,210.7 m^3 [584,000 gallons] for the RWSP normal liquid volume, defined as the water volume at the 96 percent water level excluding water below the 0 percent level, which is practically identical to the value used in the mixing analysis.

Also, US-APWR DCD Tier 2 Table 6.1-3, "Water Chemistry Specifications of the RWSP," provides the RWSP minimum boric acid concentration as 4,000 ppm. The staff agrees that applying the limiting values for the RWSP initial volume and boron concentration in the applied approach is conservative, as this reduces the calculated initial boron concentration in the RPV lower plenum prior to mixing. As documented in Table RAI 15.06.05-91.2 in the RAI response, the mixed boron concentration for the BOC case was calculated [
(Proprietary information withheld under 10 CFR 2.390)

] above the recriticality limit. The last result is documented in Table RAI 15.06.05-90.2 in the response to RAI 719-5352, Question 15.6.5-90. The applicant also pointed out mixing tests at the University of Maryland 2x4 Thermal-Hydraulic Loop and at the Rossendorf Coolant Mixing Model (ROCOM) with pump startup indicating that mixing, in addition to the lower plenum, can also occur in the reactor downcomer. To illustrate the sensitivity of the resulting mixed boron concentration to the assumed fraction of the combined lower plenum and downcomer volumes that participates in the mixing process, results from two BOC sensitivity cases were presented in Table RAI 15.06.05-91.2 in the RAI response. As the lower plenum volume represents [(Proprietary information withheld under 10 CFR 2.390)] and the downcomer, this fraction represents the base case analyzed in the RAI response. In the first sensitivity case, the coolant volume in the RPV assumed to mix with the deborated condensate volume was set at 60 percent of the combined lower plenum and downcomer volume thus crediting a portion of the downcomer volume in the mixing. In the second sensitivity case, the coolant volume in the RPV assumed to mix with the deborated condensate volume was set at 40 percent of the combined lower plenum and downcomer volume thus crediting only 79.6 percent of the lower plenum volume in the mixing calculation. [
(Proprietary information withheld under 10 CFR 2.390)
]. The applicant also explained that the analysis assumption of deborated slugs entering the reactor vessel simultaneously from all four loops was conservative. To illustrate the sensitivity of the resulting mixed boron concentration to the number of loops assumed to experience a simultaneous resumption of natural circulation flow, results from three BOC sensitivity cases assuming restart in three, two, and a single loop only were presented in Table RAI 15.06.05-91.3 in the RAI response. [
(Proprietary information withheld under 10 CFR 2.390)]. As available experimental data suggests that natural circulation does not start simultaneously in all reactor loops and, taking into account that, as stated in the response to RAI 718-5402, Question 15.6.5-83, the US-APWR emergency procedure does not recommend a concurrent restart of multiple RCPs, the staff agrees that this assumption is conservative as it maximizes the volume of clean condensate that is transported towards the core in the dilution transient. Although the staff recognizes the conservative aspects within the frame of the implemented analytical mixing approach, the acceptability of the assessment results remains open based on the following considerations. The staff finds questionable the implemented modeling formula for calculating the minimum core entry boron concentration as being equal to the initial boron concentration in the reactor lower plenum multiplied by the ratio of the lower plenum volume to the combined volume of the lower plenum and the deborated condensate as shown in Table RAI-15.06.05-90.1 in the response to RAI 719-5352, Question 15.6.5-90. Also, in assessing the amount of condensate in the loops, possible accumulation of boron-depleted liquid in the SG outlet plena was not considered in the RAI response despite available experimental evidence in this regard. Hence followup to RAI 719-5352, Question 15.6.5-91, was issued.

Open Item 15.6.5-9

RAI 861-6062, Question 15.6.5-98 (Follow-up RAI 761-5352, Question 15.6.5-91):

1. In Table RAI-15.06.05-90.1 of UAP-HF-11106, the bounding boron concentration for the lower plenum coolant prior to deborated condensate water entering the lower plenum is given as [(Proprietary information withheld under 10 CFR 2.390)

   ] identified as the maximum deborated water volume cases, demonstrate that the entire RWSP volume will be injected into the RCS prior to re-establishment of natural circulation conditions. Provide total water mass (liquid and vapor) out the break, total water mass and volume injected from the RWSP and accumulators between the start of reflux condensation to resumption of natural circulation. If the entire RWSP volume of water is not capable of being injected and mixed with the initial RCS boron concentration how is the RPV boric acid concentration determined at the time of natural recirculation resumption?

2. Figures RAI 15.06.05-85.5 and RAI 15.06.05-85.6 in the response to RAI 718-5402, Question 15.6.5-85, provide the amount of condensate generated in the SG U-tubes [(Proprietary information withheld under 10 CFR 2.390)]. The amount of condensate generated by reflux condensation depends on the break size and the number of SGs receiving emergency feedwater and can be considerably larger than the volume of the loop seals. Assess the impact and provide the analysis results that account for generation and possible RCS condensate accumulation [(Proprietary information withheld under 10 CFR 2.390)] as considered in the recriticality consequences evaluation in the responses to RAI 719-5352, Questions 15.6.5-90 and 15.6.5-91?

3. The initial RCS coolant boron concentration was determined at [(Proprietary information withheld under 10 CFR 2.390)

   ] If not, why is one not needed?

4. In Table RAI-15.06.05-90.1 of UAP-HF-11106 the initial HFP, ARO boron concentration of [(Proprietary information withheld under 10 CFR 2.390)

   ] is the minimum margin condition? If not please justify.

5. What is the basis for the applied formula for calculating the boron concentration of coolant entering the core following deborated slug insertion that is provided in Table RAI-15.06.05-90.1 in the response to RAI 719-5352, Question 15.6.5-90? Is there experimental evidence that supports the appropriateness of this simplistic equation for the purposes of the analysis? Compare the formula predictions against test data

and demonstrate that the formula, if and as applied in the US-APWR boron dilution analysis, leads to conservative results.

15.6.5.4.3.4.3 Long Term Cooling and Boron Precipitation Prevention

In the US-APWR DCD Tier 2 Section 15.6.5 analysis of boron precipitation prevention, operator actions were credited to demonstrate post-LOCA long-term core cooling. The main objective of the post-LOCA long-term cooling evaluation was to determine the switchover time from RPV downcomer injection via the DVI mode to the simultaneous RPV downcomer and hot leg injection mode to prevent the concentration of boric acid from reaching the solubility limit in the core. The staff reviewed pertinent portions of US-APWR DCD Tier 2 Section 15.6.5 and identified specific items that required additional information from the applicant. To address these outstanding items, the staff issued RAI 352-2369, Questions 15.6.5-44 through 15.6.5-55. MHI responded to these RAI questions in [Reference 15-7] and the responses are reviewed below.

In RAI 352-2369, Question 15.6.5-44, the staff recognized that the US-APWR post-LOCA, long-term cooling evaluation model was described in the US-APWR DCD Tier 2, Revision 3, as similar to models that had been previously used and questioned if the US-APWR evaluation model has been previously reviewed by the NRC. The staff also questioned if a code manual exists for the model. The applicant responded [Reference 15-7] that the long-term cooling model has not been reviewed by the NRC. It was also stated in the response to RAI 352-2369, Question 15.6.5-44, that the detailed description of the model was provided in Appendix B, “Post-LOCA Long Term Cooling Evaluation Model for US-APWR” in [Reference 15-7]. Based on the review findings related to Appendix B in [Reference 15-7], the staff identified the following Open Item.

Open Item 15.6.5-10

RAI 861-6062, Question 15.6.5-92 (Follow-up to RAI 352-2369, Question 15.6.5-44):

> US-APWR DCD Tier 2 Subsection 15.6.5.3.1.3 “Post-LOCA Long Term Cooling Evaluation Model” describes the model that is used to predict the boric acid concentration in the reactor core during the LOCA long-term cooling period. A more detailed description of the model is provided in Appendix B to UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1” as part of the response to RAI Question 15.6.5-44. Identify the decay heat model used in the US-APWR boron precipitation analyses performed with the long-term cooling evaluation model. Provide the decay heat multiplier assumed in the calculations. The amount of liquid in the mixing volume depends on the predicted vapor volumetric fraction within this volume. The applied axial power profile can impact the volumetric vapor fraction in the core region. If the US-APWR boron precipitation analyses were not produced with a limiting bottom-peaked axial power profile, provide the impact on the precipitation timing resulting from an assumed limiting bottom-peaked axial power shape. Also discuss the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79.

In RAI 352-2369, Question 15.6.5-45, the staff questioned how the safety injection is switched from direct vessel injection to both vessel and hot leg injection. The staff also asked for an explanation of the manual switchover procedure, including what parameter prompts the switchover and references for relevant operator procedures. The applicant responded in [Reference 15-7] that the switchover occurs around four hours after the time the operators

recognize a LOCA has occurred. Simultaneous vessel and hot leg injection is established by closing some (but not all) operating DVI line isolation valves and opening the associated hot leg injection isolation vales. These remote valves are manually operated from the main control room. These operator action procedures will be written in the future as part of the emergency procedure guidelines.

Although the response to RAI 352-2369, Question 15.6.5-45 was found acceptable, the staff found it necessary to further clarify if the timing of switchover to simultaneous vessel and hot leg injection is established relative to the start of reflood, as assumed in the post-LOCA, long-term cooling evaluation model, relative to the break initiation or relative to the time the operators recognize a LOCA has occurred. In addition, the staff considered that a specific time delay limit needed to be established by the applicant. To address these findings, the staff formulated an RAI question tracked below as an open item.

Open Item 15.6.5-11

RAI 861-6062, Question 15.6.5-93 (Follow-up to RAI 352-2369, Question 15.6.5-45):

> According to Appendix B, "Post-LOCA Long Term Cooling Evaluation Model for US-APWR," to the RAI 352-2369, Question 15.6.5-44, response, the boric acid precipitation calculation is initiated at the beginning of the reflood phase to determine the timing of boric acid precipitation. However, the response to RAI 352-2369, Question 15.6.5-45, refers to the point in time when the operators recognize a LOCA to characterize the timing of manual switchover to hot leg injection and states that the related operator action procedures will be provided in future Emergency Procedure Guidelines (EPGs). Explain how it will be ensured in the EPGs that the timing of such manual switchover to hot leg injection will be defined so that it occurs prior to the point of boric acid precipitation as predicted by post-LOCA, long-term cooling evaluation model.

The long-term cooling analysis evaluation model uses numerous equations, which are provided in Section 15.6.5.3.1.3, "Post-LOCA Long term Cooling Evaluation Model," of the US-APWR DCD Tier 2 Revision 3 [Reference 15-46]. In RAI 352-2369, Question 15.6.5-46, the staff questioned how these equations are derived and how they are solved. As an example, the spill flow shown in Figure 15.6.5-41 of the US-APWR DCD Tier 2 Revision 3 [Reference 15-46, p. 15.6-146] is not shown in the equations; there is no information on how the ratio of RV injection flow and hot leg injection flow is determined; no equations are shown for the calculation of the void fraction and how it is entered into the equation (other than that it is calculated by Yeh's correlation), etc. The applicant responded in [Reference 15-7] that a detailed description of the evaluation model, including equations and calculation methods, was included in the response to RAI 352-2369, Question 15.6.5-44. It was also noted that the spill flow is treated as a part of RWSP volume and therefore is not shown in the equations because it circulates in the RWSP and does not directly enter into the mixing volume. The staff found the response to RAI 352-2369, Question 15.6.5-46, acceptable as the applicant explained that the information was provided in Appendix B, "Post-LOCA Long Term Cooling Evaluation Model for US-APWR" in [Reference 15-7] as part of the response to RAI 352-2369, Question 15.6.5-44.

In RAI 352-2369, Question 15.6.5-47, the staff questioned whether the equations are solved analytically or numerically and asked for an explanation of the solution procedures. In the response to this question [Reference 15-7], the applicant explained that the equations are solved numerically at each timestep and referred again to the response to RAI 352-2369, Question 15.6.5-44, for more detailed description of the evaluation model. The staff found the

response to RAI 352-2369, Question 15.6.5-47, acceptable as the applicant answered the question and referred to Appendix B, “Post-LOCA Long Term Cooling Evaluation Model for US-APWR,” in the response to RAI 352-2369, Question 15.6.5-44 [Reference 15-7] for detailed description of the model.

The size of the mixing volume is controlled by the external loop resistance and the balance of hydrostatic heads between the downcomer and inner vessel regions. In RAI 352-2369, Question 15.6.5-48, the staff asked how the evaluation model accounts for these effects and how the mixing volume makeup flow rate is determined as a function of time. The applicant responded in [Reference 15-7] that in the post-LOCA, long-term cooling evaluation model, mixing volume makeup flow rate is the sum of the core evaporation rate and the change rate of mixing volume water mass. The mixing volume includes the upper plenum below the hot leg bottom elevation. In practice, the core side mixture level is considered higher than that elevation; thus the range of mixing volume is considered conservative. In the post-LOCA, long-term cooling period, the downcomer water level is maintained more than the cold-leg bottom elevation by HHIS. It is unclear how the downcomer level can be higher than the cold-leg bottom elevation, specifically for a large, cold-leg break. On the other hand, HHIS water flowing into the core evaporates due to core decay heat and core side mixture level swells. Downcomer side and core side hydraulic heads were calculated. Table-1, “Hydraulic Heads Calculation Condition,” in the response to RAI 352-2369, Question 15.6.5-48, shows calculation conditions of these hydraulic heads. Void fraction was calculated using the modified Yeh correlation. Figure-1, “Hydraulic Head Transient during Post-LOCA Long-Term Cooling,” in the same RAI response shows the calculated time-history of hydraulic heads in post-LOCA, long-term cooling. The core-side hydraulic head is less than the downcomer head because there are many steam bubbles in the core and the upper plenum. The hydraulic head at the time of hot-leg switch-over (4 hours) becomes 53 percent of downcomer head. Steam generated in the core flows through the hot leg, the SG, the RCP, and out through the break which is located at the cold leg. In consideration of steam state change accompanying the following phenomena, the external loop was divided into nodes to calculate flow resistance. Flow resistance of each node was calculated, and total loop flow resistance was estimated.

While the method for computing the mixing volume makeup flow rate was noted in the response, it is not clear that this method is consistent with the pressure difference between the core and downcomer. For example, the core average void fraction is determined from the Yeh correlation. From that, the total liquid volume in the core is computed. Based on the change in core liquid volume and the core steaming rate, the makeup flow rate is computed. To address the above findings, the staff formulated an RAI question tracked below as an Open Item.

Open Item 15.6.5-12

RAI 861-6062, Question 15.6.5-94 (Follow-up to RAI 352-2369, Question 15.6.5-48):

> In response to RAI 352-2369, Question 15.6.5-48, it was stated that the mixing volume makeup flow rate is the sum of core evaporation rate and the change rate of the mixing volume water mass. Compare the calculated makeup flow rate using this method with the flow rate that would be obtained from the pressure difference between the downcomer and core. Also compare this flow rate to the total HHIS flow rate. It was also stated in the original RAI response that “In the post-LOCA long-term cooling period, downcomer water level is maintained more than cold leg bottom elevation by HHIS.” Explain how the downcomer water level can be above the cold leg bottom for a large cold leg break (double ended guillotine break).

In RAI 352-2369, Question 15.6.5-49 [Reference 15-15], the staff questioned how the initial boric acid concentration of the mixing volume (at the beginning of the reflood phase) was determined.

The staff also asked for a table showing the initial values of the variables appearing in the equations for typical large and small break LOCAs along with a discussion of the boundary variables and how they are determined. The applicant responded [Reference 15-7] that the initial boric acid concentration of the mixing volume was conservatively assumed to be the maximum concentration of the accumulators [(Proprietary information withheld under 10 CFR 2.390)]. The staff found the response to RAI 352-2369, Question 15.6.5-49, acceptable as the applicant provided the requested information.

In RAI 352-2369, Question 15.6.5-50, the staff asked how the boric acid from the accumulators is accounted for in the evaluation model since it is not included in the equations. The applicant responded in [Reference 15-7] that some of the boric acid from the accumulators goes to the core region and some spills out into the RWSP through the break. The boric acid from the accumulators is split between the two control volumes. It was also stated that a calculation of the initial mass of the mixing volume and RWSP is described in the response to RAI 352-2369, Question 15.6.6-49 [Reference 15-7]. The staff found the response to RAI 352-2369, Question 15.6.5-50, acceptable as the applicant provided the requested information.

In the US-APWR DCD Tier 2, the applicant uses a boron concentration limit of 29.27 wt. %, which is the precipitation concentration at atmospheric pressure. In RAI 352-2369, Question 15.6.5-51, the staff questioned the uncertainty of the assumed limit in the evaluation model. The staff also asked about margin for operator error. The applicant responded in [Reference 15-7] that the limit was taken from the boric acid solubility described in US-APWR DCD Tier 2 Reference 15.6-28 and that no uncertainty was added. The applicant stated that the concentration limit at atmospheric pressure is conservative since the actual pressure in the core would be at least several psi higher at the time of the switchover to simultaneous DVI and hot leg injection. A 3.0 psi increase in pressure results in a solubility limit of 32 wt. %. The evaluation model shows that it takes about five and a half hours after the LOCA for the concentration to reach this limit. Given that the operators make the switchover around four hours, there is one and a half hours of margin. The staff found the response to RAI 352-2369, Question 15.6.5-51, acceptable as the applicant demonstrated that there is conservatism in the applied boric acid precipitation limit related to the mixing volume pressure being assumed at the lowest possible atmospheric level for the purposes of the boron precipitation analysis. In addition, the applicant showed that there are one and a half hours of a corresponding margin for the operator action. The staff agrees with the conclusion reached by the applicant that this margin is enough to recover from operator error.

The evaluation model assumes that all the vapor flow out of the mixing volume is returned to the RWSP. However some of the vapor in the containment may not return to the RWSP, thus potentially increasing the boric acid concentration. In RAI 352-2369, Question 15.6.5-52, the staff asked for a discussion of potential impacts of this on the conclusion that the boric acid concentration would stay below the limiting value (29.27 wt. %), in view of the results which show that the concentration is within 3 percent of the limiting value before the switchover for some transients. The applicant responded in [Reference 15-7] that the vapor content in the containment is included in the RWSP volume because the amount is estimated to be small and

does not affect the evaluation. In order to confirm the impacts of the vapor, the applicant performed a sensitivity analysis. As a result of the sensitivity calculation, the amount of vapor was calculated to be about 4 percent of the total boric acid solution mass in the RWSP, accumulator, and RCS. The difference of the boric acid concentration four hours after the LOCA occurred was calculated to increase 1.0 wt. percent.

This result indicates that the impact of the vapor amount in the containment would be so small that the impact would fall below the 3 percent margin between the analysis result of the boric acid in the DCD and precipitation limit (29.27 wt. %). The staff found the response to RAI 352-2369, Question 15.6.5-52, acceptable as the applicant showed that accounting for the conservatively assessed vapor content within the containment had a relatively small impact on the available margin to the applied precipitation limit.

During the post-LOCA cooling period, containment spray may be activated, which is not accounted for in the equations used in the evaluation model for boron precipitation. In RAI 352-2369, Question 15.6.5-53, the staff asked for a discussion on the impact of the containment spray on the boric acid concentration. The applicant responded in [Reference 15-7] that the containment spray takes water from the RWSP and releases it into the upper containment. The sprayed water is then collected back in the RWSP. The water in the containment spray is assumed to be a part of the RWSP water. Therefore, the containment spray is not shown in equations described in the US-APWR DCD. The applicant also explained that the containment spray system moves RWSP water to the “non-available water volume” in the containment. In the evaluation model, RWSP volume includes these “non-available water volumes” as a boric acid water source. There are two major types of non-available water volumes: one is “return water on the way to RWSP,” which is temporarily separated but eventually returns to the RWSP, while the other is “ineffective pool,” in which the water never returns to the RWSP. Return water will have the same boric acid concentration as the circulated water in the RWSP in the post-LOCA, long-term cooling; thus, it is reasonable to include “return water on the way to RWSP” to the RWSP volume. Ineffective pool water is not thought to be mixed with circulated water in the mixing volume or the RWSP. Though ineffective pool water is eliminated from the RWSP volume in the post-LOCA, long-term cooling evaluation, the boric acid concentration of the mixing volume at the time of hot leg switchover does not change or become lower because the sum of boric acid mass in the RWSP decreases. In its response, the applicant identified two effects of ineffective pool water on the boric acid concentration in the RWSP: (1) Initial boric acid concentration in the RWSP volume will increase since the RWSP water would not mix with the RCS water, which has lower boric acid concentration than RWSP water. In the case where the RWSP water will not mix with RCS water, boric acid concentration of the circulated water will increase; (2) In general, the water that is evaporated in the core will be condensed by the containment spray and returned as deborated condensate to the RWSP thus causing the boric acid concentration in the RWSP to decrease. The induced rate of reduction of the boric acid concentration in the RWSP will be higher at a decreased RWSP liquid inventory as the produced condensate will be available to deborate a smaller amount of RWSP water. To confirm these effects, the applicant performed a sensitivity analysis for the large break LOCA base case considered in the DCD accounting for the ineffective pool water. The calculated time-history of the mixing volume boric acid concentration presented in the response showed no significant difference between the sensitivity case and the DCD base case results due to the compensating nature of both effects. The staff found the response to RAI 352-2369, Question 15.6.5-53, acceptable as the applicant showed that accounting for the containment spray operation had little impact on the available margin to the applied boron precipitation limit.

A certain concentration of boric acid is required to keep the core subcritical after scram following a LOCA. In RAI 352-2369, Question 15.6.5-54, the staff asked the applicant to specify what minimum boric acid concentration is required to keep the core subcritical after scram and indicate it in Figures 15.6.5-42 and 15.6.5-43 of the US-APWR DCD Tier 2 Revision 3 [Reference 15-46, pp. 15.6-147 and 15.6-148].

MHI responded in [Reference 15-7] that according to the criticality calculation, the boric acid concentration should be maintained over 1.4 wt. % (2,450 ppm) to keep the core subcritical in the long-term period after a LOCA. The applicant also provided plots, which show DCD Figures 15.6.5-42, "US-APWR Post LOCA Long Term Cooling Evaluation for 14.7 psia" and 15.6.5-43, "US-APWR Post LOCA Long Term Cooling Evaluation for 120 psia" with the added minimum boron concentration noted. After the switchover to the simultaneous DVI and hot leg injection mode, the boric acid concentration in the mixing volume decreases, but does not fall below the lower limit of 1.4 wt. %. The staff found the response to RAI 352-2369, Question 15.6.5-54, acceptable as the applicant showed that the reactor remains subcritical following the switchover to the simultaneous DVI and hot leg injection mode.

In RAI 352-2369, Question 15.6.5-55, considered in the previous subsection of this report, the staff asked if M-RELAP5 includes a boric acid concentration option. The staff also asked if, provided that such an option was available, it was activated in the SBLOCA analysis. In its response to RAI 352-2369, Question 15.6.5-55 [Reference 15-7], the applicant explained that while both M-RELAP5 as well as RELAP5-3D include a boron tracking capability, the function was not used in the US-APWR SBLOCA analyses. The staff found the response to RAI 352-2369, Question 15.6.5-55, acceptable as the applicant relied on a separate evaluation model to analyze long-term cooling boron precipitation for both small- and large-break LOCAs.

As part of the review of the boron precipitation evaluation model, described in Appendix B, "Post-LOCA Long Term Cooling Evaluation Model for US-APWR" in [Reference 15-7], the staff identified specific items that required additional information from the applicant. To address these outstanding items, the staff issued RAI 706-5339, Questions 15.6.5-81 and 15.6.5-82 [Reference 15-35] as well as RAI 719-5352, Questions 15.6.5-87, 15.6.5-88, and 15.6.5-89 [Reference 15-39].

US-APWR DCD Tier 2 Revision 3 Subsection 15.6.5.3.1.3 "Post-LOCA Long-Term Cooling Evaluation Model" describes the model that is used to predict the boric acid concentration in the reactor core during the LOCA, long-term cooling period. A more detailed description of the model is provided in Appendix B to UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1," July 2009, [Reference 15-7] as part of the response to RAI 352-2369, Question 15.6.5-44. According to this response, the evaluation model has not been previously reviewed by the U.S. NRC.

In the provided US-APWR boron precipitation analysis, the control mixing volume for calculating the boric acid concentration in the core includes the volume of the following regions: (1) core region volume, (2) upper plenum volume below the hot leg bottom elevation, and (3) one half of the lower plenum volume.

The average boric acid concentration computed with the above defined mixing volume is based on the assumption that 50 percent of the liquid in the lower plenum mixes homogeneously and instantly with the liquid content of the two remaining regions of the mixing volume. At the same time, it is recognized that if the density of the colder liquid residing in the lower plenum is higher than the coolant density in the core region, thermal stratification can preclude mixing between

the regions. According to US-APWR DCD Tier 2 Table 4.4-4, one half of the lower plenum volume corresponds to 61 percent of the entire active core fluid volume. As such, inclusion of this lower plenum portion in the control mixing volume can significantly affect the predicted boric acid buildup. On March 1, 2011, RAI 706-5339, Question 15.6.5-81, the staff asked the applicant to provide the technical basis in support of the proposed inclusion of 50 percent of the lower plenum volume in the control mixing volume for the US-APWR boric acid precipitation analysis.

In addition, the staff asked the applicant to demonstrate the applicability of test data used as part of the technical basis to the US-APWR reactor design by describing the applicable scaling methodology along with the scaling results for the US-APWR design.

On April 28, 2011, the applicant responded to RAI 706-5339, Question 15.6.5-81 [Reference 15-43], and explained that half of the lower plenum was assumed to be included in the mixing volume, which was equivalent to the entire lower plenum volume being subjected to half of the core boric acid concentration. In the response, the applicant presented experimental test results obtained at the “BACCHUS” test facility located at the MHI Takasago Research and Development Center. The test facility, which is representative of a Japanese 3-loop PWR plant, was used to simulate the boric acid behavior inside the reactor pressure vessel during the post-LOCA long-term core cooling phase. The test vessel is represented in slab geometry to capture two-dimensional fluid behavior with respect to the radial and axial directions within the reactor vessel. Circumferential flow effects were expected to be negligible during the post-LOCA long-term cooling phase due to the relatively low liquid flow velocity in the reactor vessel downcomer (less than 0.5 cm/s [0.2 in/s] two hours after the LOCA occurs). The test rig is full-scale in height and nearly full-scale in the radial direction and uses axial symmetry to represent only the area from the core center to the downcomer. The slab width along the circumferential direction is based on the fuel assembly scale and amounts to 15.2 cm [0.5 ft]. The volumetric scale with regard to the reference plant is approximately 1:80. Electrical heaters are used to simulate the fuel decay heat. The heaters provide uniform axial power distribution with the inner third of the core heaters powered at 150 percent of the average power level, the middle third powered at the average power, and the peripheral one third having 50 percent of the average power density. Instrumentation is installed in the test vessel to measure pressure, fluid temperature and boron concentration. The boric acid concentration was obtained by measuring the electrical conductivity of the fluid. In the test, all electrical heaters were powered simultaneously to boil the water in the core with the steam leaving the test vessel and entering into a steam-water separator tank. As the test rig simulated power of 270 kW was relatively low compared to the reference plant, the initial boric acid concentration in the test vessel and the boric acid concentration of the injected water were increased to 5.15 wt. % to accelerate the boric acid accumulation rate in the test vessel. The temperature of the injected ECCS water was set at 60 C [140 °F] as a representative sump water temperature. As described in the RAI response, the ratio of the lower plenum volume to the core volume in the BACCHUS test rig is higher in comparison to the US-APWR (75.9 percent versus 69.1 percent, correspondingly). The staff agrees with the applicant’s statement that due to this geometrical discrepancy the influence of lower plenum mixing is relatively smaller in the BACCHUS test facility when compared to US-APWR in terms of the participating lower plenum mixing fraction. At the same time, the elevation difference between the lower plenum bottom elevation and the core bottom elevation in the BACCHUS facility, although being smaller in comparison to US-APWR, remains very close to the prototype (2.755 m [9.04 ft] versus 2.990 m [9.81 ft], correspondingly). With regard to additional important geometrical factors, the staff finds that the influence of lower plenum mixing is somewhat higher in the BACCHUS test facility when compared to US-APWR in terms of the participating lower plenum mixing fraction. Based on the above considerations with

regard to the geometrical similarity between the BACCHUS test facility and US-APWR, the staff agrees with the applicant's assertion that the boric acid mixing behavior between the core and the lower plenum is consistent between the US-APWR and the BACCHUS test facility so that the lower plenum mixing fraction obtained from the BACCHUS boric acid mixing test is applicable to the US-APWR evaluation. At the same time, the staff recognizes that the temporal characteristics of the mixing process obtained in the BACCHUS test rig are not representative of the US-APWR prototype due to the relatively low core power simulated in the test.

Figure 26 in the RAI response shows the time variation of the lower plenum mixing fraction determined from the ratio of the volume-averaged boric acid concentrations in the core and lower plenum. The time dependent volume-averaged concentrations were calculated from the local concentration measurements. The figure shows that the lower plenum mixing fraction increases with time and reaches 50 percent in approximately 4.6 hours. What is of direct relevance to the US-APWR is Figure 27 which shows the variation of the observed lower plenum mixing fraction as a function of the volume-averaged core boric acid concentration. As seen from the figure, the lower plenum mixing fraction reaches 50 percent when the core boric acid concentration increases to approximately 19 wt. %. Based on this experimental observation from the BACCHUS test, the staff agrees that it is reasonable to include 50 percent of the lower plenum volume in the calculated mixing volume. In accepting the applicant's conclusion that the validity of this assumption was demonstrated by the BACCHUS boric acid mixing test data, the staff took into consideration that the boric acid precipitation criterion of 29.27 wt. % defined in US-APWR DCD Tier 2 Section 15.6.5.3.1.3, "Post-LOCA Long Term Cooling Evaluation Model," exceeds, by a large margin of about 10 wt. %, the experimentally observed core boric acid threshold concentration of approximately 19 wt. % for crediting 50% of the lower plenum volume in the mixing volume. Taking into consideration this margin, the staff finds the existing geometrical disparities between the BACCHUS test facility and the prototype as well as possible effects associated with the applied test conditions, including the ECCS injection temperature, acceptable. Thus, the staff finds the response to RAI 706-5339, Question 15.6.5-81, acceptable and the RAI question resolved.

In the US-APWR boron precipitation analysis, consideration of fluid mixing between coolant in areas of the reactor lower plenum and such residing in adjacent reactor core regions would require conditions under which the boric acid solution in these core regions becomes denser than the coolant in the lower plenum. Under such conditions, participation of a certain fraction of the lower plenum in the control mixing volume can be considered for crediting in the US-APWR boric acid precipitation evaluation.

On March 1, 2011, RAI 706-5339, Question 15.6.5-82, the staff asked the applicant to provide a conservative assessment of the expected coolant temperature in the reactor vessel lower plenum region during the post-LOCA, long-term cooling phase along with a list of all relevant assumptions under the most limiting LOCA conditions. The staff requested that the applicant provide the time period and conditions under which lower plenum participation in the control mixing volume can be considered for crediting in analyzing the effectiveness of the hot leg switchover to avoid boron precipitation. In addition, the applicant should : (1) show the lower plenum liquid density based on the provided temperature and any other contributing factors if applicable, (2) provide the density of the liquid in the core based on corresponding coolant temperatures, boric acid concentrations, and any other contributing factors along with a list of all relevant assumptions, (3) explain how the representative core liquid density conservatively accounts for possible boric acid concentration variations within the reactor core, (4) present comparison plots for the representative coolant temperatures, boric acid concentrations, and liquid densities attributed to the lower plenum and core regions as functions of time, (5) based

on the performed assessments, provide the time period and conditions under which lower plenum participation in the control mixing volume can be considered for crediting in analyzing the effectiveness of the hot leg switchover to avoid boron precipitation, (6) and, as appropriate, show sensitivity analyses to support the conclusions and include an assessment of the uncertainties associated with the main contributing parameters.

On April 28, 2011, the applicant responded to RAI 706-5339, Question 15.6.5-82 [Reference 15-43], that density stratification between the saturated core liquid and the colder water in the lower plenum becomes unstable when the liquid density in the core exceeds the liquid density in the lower plenum. As a result, gravity-driven fluid mixing between the core and lower plenum regions takes place. In the analysis presented in the response, the applicant conservatively assumed that the lower plenum water temperature was at the lowest plausible value of 3.9 C [39 °F] and the core liquid was at saturation at the minimum possible pressure of one atmosphere (100 C [212 °F]). The boric acid concentrations in the core and in the lower plenum were set at the initial value of 2.402 wt. % thus forming a stable stratified configuration due to the density difference at the assumed liquid temperatures in both volumes. Using an available correlation that accounts for the effect of boric acid concentration on the water density, it was determined that the boric acid concentration in the saturated core liquid at which the water in the core region would be equal to the density of the lower plenum water was 15.690 wt. %. At higher core concentrations, gravity-driven mixing between the volumes is expected to take place.

Next, the applicant performed a quasi-steady-state mass and energy balance for the lower plenum region to determine the mixing flow rate between the core and the lower plenum as well as the boric acid concentrations in both regions following the incipience of fluid mixing between the regions. The following assumptions were used in the calculation: (1) core boiloff flow rate matched Appendix K decay heat as the fluid entering the core from the lower plenum is at saturation; (2) water initially present in the lower plenum was from ECC injection and has a boric acid concentration of 2.402 wt. %; (3) ECC flow rate into the lower plenum was equal to the core boiloff rate; (4) ECC water temperature was conservatively assumed to be at the lowest plausible value of 3.9 C [39 °F]; (5) lower plenum boric acid concentration was determined by the ECC flow from the RWSP and the mixing flow from the core. After the core concentration reached the critical value of 15.690 wt. %, at which point mixing between the regions starts, the lower plenum boric acid concentration was calculated from the condition that the liquid density in the lower plenum was equal to the core liquid density. Once the predicted boric acid concentration in the lower plenum reached one half of the core concentration, the lower plenum was maintained at half the value of the core boric acid concentration consistent with the assumption in US-APWR DCD Tier 2 Section 15.6.5.3.1.3, “Post-LOCA Long Term Cooling Evaluation Model,” that 50% of the lower plenum volume participates in the mixing volume. The results were presented in Figure 1, “Boric Acid Concentration during post-LOCA Long-term Cooling,” and in Figure 2, “Liquid Density during post-LOCA Long-term Cooling,” in the response to RAI 706-5339, Question 15.6.5-82 [Reference 15-43] showing the predicted boric acid concentrations and the liquid densities in both participating regions, respectively. The results demonstrated that the liquid density in the core reached the lower plenum density in approximately 30 minutes. After approximately 130 minutes, the lower plenum boric acid concentration reached half of the core boric acid concentration, which is equivalent to crediting half of the lower plenum volume in the core mixing volume. Following this point, the predicted core boric acid concentration is identical with the boron precipitation result shown in US-APWR DCD Tier 2 Figure 15.6.5-42, “US-APWR Post LOCA Long Term Cooling Evaluation for 14.7 psia,” in which analysis mixing between the lower plenum and the core is assumed to begin from the onset of the transient. This is also seen in Figures 5 and 6 provided in the RAI

response, which show the calculated time history of the boric acid concentration and the liquid densities in the core and the lower plenum with the initial lower plenum liquid temperature and ECC water temperature at saturation as assumed in the US-APWR DCD Tier 2 analysis result shown in US-APWR DCD Tier 2 Figure 15.6.5-42.

The results show that the boric acid content in the core liquid is predicted to reach the precipitation limit at about 280 minutes [4.69 hours] with a significant delay of about 150 minutes after the point in time when 50 percent of the lower plenum volume can be credited in determining the boron mixing volume. Based on this outcome and considering the assumed initial lower plenum and ECC water temperatures that conservatively delay the start of mixing between the lower plenum and core regions, the staff agrees with the applicant’s conclusion that the initial lower plenum liquid temperature has no effect on the predicted time to boron precipitation as presented in US-APWR DCD Tier 2 Section 15.6.5.3.3.3, “Post-LOCA Long Term Cooling Results.” Since the time of safe switchover to hot leg injection is determined by the available time before the boric acid concentration in the core mixing volume reaches the boric acid precipitation limit, the staff finds the response to RAI 706-5339, Question 15.6.5-82, acceptable and the RAI question resolved.

Fluid mixing between coolant in the US-APWR lower plenum and in adjacent reactor core regions can take place, resulting in possible localized coolant temperature variations in the lower plenum and core inlet areas in the US-APWR boric acid precipitation analysis. Due to the strong dependence of the boric acid solubility limit on the solution temperature, precipitation can be generated by such local coolant temperature distributions in areas where colder coolant can reside. On March 17, 2011, the staff issued RAI 719-5352, Question 15.6.5-87, asking the applicant to provide a calculation for the boric acid solubility limit at a solution temperature that conservatively bounds expected coolant temperature variations in the reactor vessel lower plenum during post-LOCA, long-term cooling. The staff requested this analysis to ensure that the boric acid concentration in the lower plenum remains below the precipitation limit after the LOCA initiation. In addition, the applicant was asked to: (1) provide a plot showing the determined precipitation limit as a function of time after the LOCA initiation, (2) provide relevant data and/or equations used to compute the result as well as those used to compute any other boric acid precipitation limits applied in the US-APWR precipitation analysis, (3) list all assumptions made in calculating the precipitation limits and discuss the impact of each individual assumption on the limiting concentrations obtained, (4) if a parameter that changes in time is represented by a single value, explain how this value was computed and the point in time or time period for which it is representative, (5) if a volume average quantity is used to represent the conditions in a certain region modeled by a control volume, explain how the spatial distribution effects associated with this parameter have been accounted for in obtaining the volume average value, (5) and in considering possible effects related to time and space variations, show that the results applied led to conservative predictions.

In a May 25, 2011, response to RAI 719-5352, Question 15.6.5-87 [Reference 15-44], the applicant stated that mixing of highly borated water from the core region with the colder water in the lower plenum could result in a mixing temperature that is below the solubility limit. As the mixing process also decreases the resulting boric acid concentration, the applicant provided results from a detailed analysis to show that no boric acid precipitation will occur in the lower plenum. In the analysis presented in this response, the applicant conservatively assumed that the core liquid was saturated at the minimum possible pressure (100 °C [212 °F] at atmospheric pressure) and the lower plenum water temperature was 3.9 °C [39 °F]. The analysis was performed for assumed ratios of lower plenum water mass to core water mass ranging broadly from 0.1 to 100. In addition, the boric acid concentration of the lower plenum water was

assumed at its initial value of 2.402 wt. % and the boric acid concentration of the core fluid was set at 16.0 wt. %. The staff finds the last value acceptable, as it is in accordance with the response to RAI 706-5339, Question 15.6.5-82, reviewed in this report and showing that mixing between the core and lower plenum regions starts when the core boric acid concentration exceeds 15.69 wt. % even at a lower plenum liquid temperature of 3.9 °C [39 °F].

The analysis results were presented in Figure 2, “Liquid Temperature vs. Mixing Fraction (R_{mix}),” and in Figure 3, “Boric Acid Concentration vs. Mixing Fraction (R_{mix}), in the response to RAI 719-5352, Question 15.6.5-87 [Reference 15-44] showing the variation of the mixed water temperature and mixed water boric acid concentration, respectively. As seen from Figure 3, the predicted mixture boric acid concentration remains below the solubility limit over the entire range of the mixing fractions from 0.1 to 100. At an expected mixing ratio of approximately 0.5 for the US-APWR, Figure 3 shows a significant margin to the precipitation limit. Furthermore, the staff agrees that the assumed lower plenum initial temperature of 3.9 C [39 °F] is conservatively low as the lower plenum fluid would become warmer due to heat release from reactor vessel metal structures during the reflood and early post-reflood phase. In this regard, the robust degree of conservatism in the applicant’s analysis is demonstrated by the mixed water boric acid concentration predicted with a more realistic lower plenum temperature of 15.6 C [60 °F] and shown in Figure 4 in the response to RAI 719-5352, Question 15.6.5-87. A significant margin to the precipitation limit is seen over the entire range of mixing ratios. Therefore, based on the quantified margin to the precipitation limit and the conservatism imbedded in the presented analysis results, the staff agrees with the applicant’s conclusion that no boric acid precipitation would occur as a result of gravity driven mixing between the core and the lower plenum. The staff finds the response to RAI 719-5352, Question 15.6.5-87 acceptable and the RAI question resolved.

In the US-APWR design, a switchover from direct vessel ECCS injection mode to a simultaneous injection mode involving direct vessel and hot leg ECCS injection is used to prevent boric acid precipitation and to ensure core cooling following a LOCA. During the simultaneous injection mode, the steam flow through the reactor hot legs can cause liquid entrainment and thus impede delivery of ECCS flow into the upper plenum. In addition, liquid holdup in the hot leg horizontal and inclined sections as well as in the connected SG regions can increase the loop resistance. In turn, this will cause a corresponding increase of the upper plenum pressure thus limiting the growth of the control mixing volume. US-APWR DCD Tier 2 Section 15.6.5.3.3.3 “Post-LOCA Long Term Cooling Evaluation Results” only refers to entrainment threshold calculations as an evaluation basis for concluding that sufficient reactor core cooling is provided following the switchover to simultaneous ECCS injection after a LOCA. On March 17, 2011, the staff issued RAI 719-5352, Question 15.6.5-88, asking the applicant to describe the entrainment model and provide the results from entrainment calculations performed for the US-APWR to demonstrate that hot leg injection is capable of preventing effectively boric acid precipitation for this reactor design. In addition, the applicant was asked to (1) discuss the applicability of the selected correlations under US-APWR specific conditions; (2) list all assumptions made in the calculations including assumptions related to the decay heat model and core decay rate calculations as well as ECCS performance; (3) provide an assessment for the earliest point in time after which the liquid delivery into the upper plenum is sufficient enough to compensate for the core boil-off rate and flush the core; (4) address possible impacts of assumptions and uncertainties associated with key parameters on the critical time point obtained; (5) and present plots showing the time variation of quantities such as pressure, temperature, injected ECCS flow rate, steam flow rate, liquid flow rate, and entrainment rate as used and obtained in the analysis.

On May 19, 2011, the applicant responded to RAI 719-5352, Question 15.6.5-88, [Reference 15-41] explaining that an operator action is credited to switch the operating DVI lines to the hot leg injection line in a simultaneous RV and hot leg injection mode to prevent boric acid precipitation following a LOCA.

It was stated that although two of the four SI injection lines would be switched for simultaneous injection in practice, it was assumed that only one injection line was used for hot leg injection in the US-APWR LOCA safety analysis since one SI is assumed to fail and one SI is assumed to be unavailable due to maintenance activities. It was stated that the earliest hot leg switchover time was determined from the following three criteria: (1) the time when the hot leg steam velocity drops below the entrainment threshold value; (2) the time when adequate ECC injection flow can be delivered to the reactor vessel in counter-current flow in the hot leg; and (3) the time when hot leg injection flow exceeds core boiloff flow and can dilute the boric acid concentration in the core.

In assessing the liquid entrainment threshold in hot leg, the applicant used the Ishii-Grolmes liquid entrainment onset criterion explaining that the applied entrainment correlation is valid for flow conditions under which the liquid phase does not occupy a significant portion of the pipe as expected in the hot legs in the post-LOCA phase and viscous effects in the liquid phase are not dominant, i.e. the liquid phase is in a turbulent regime. In the analysis, the fluid properties were taken at atmospheric conditions (101.3 kPa [14.7 psia]) when computing the threshold vapor superficial velocity and using it to determine the core steaming mass flow rate. In computing the last quantity it was also assumed that all four hot legs were drained and vented steam equally. Then, the core decay heat power was used to conservatively determine the core steaming rate by assuming that the ECC water entering the core was at saturation so that the entire decay power went to steam production. The applicant assumed a corrected nominal initial core power of 4,540 MWt accounting for 2 percent measurement uncertainty (4,451 MWt x 1.02 = 4,540 MWt) to assess the core decay power fraction and used tabulated data for the decay power fractions versus time to determine the point in time when the hot leg steam velocity reached the assessed liquid entrainment threshold condition in the hot leg. It was stated that the tabulated decay heat power fractions, provided in Table 15.06.05-88-01 in the RAI response were in accordance with 10 CFR 50 Appendix K requirements assuming decay heat of 1.2 times the values for infinite operating time based on the ANS 1971 Decay Heat Standard. Thus, it was determined that the steam flow in the hot legs should drop below the assessed entrainment threshold at about 5,400 seconds after the reactor trip.

The CCFL condition in hot leg was analyzed to determine the time when sufficient injection flow can reach the reactor vessel so that the safety injection flow can flush the core. For this purpose, the applicant used a CCFL correlation based on the Kutateladze number and referred to Section 8.1.5 in revision 2 of MUAP-07013-P, "Small Break LOCA Methodology for US-APWR." It was assumed in the analysis that only one hot leg injection line was active and that the core boil-off steam was vented through all four loops equally. Using the CCFL correlation, the core steaming rate when the hot leg injection liquid flow rate balances the core boil-off rate was determined. Then, the corresponding core decay power and time after the reactor trip were calculated in the same manner as applied in assessing the entrainment threshold criterion considered above. It was found that at about 4,560 seconds after the reactor trips, the safety injection flow can reach the reactor vessel without being impeded due to CCFL taking place in the hot leg.

The criterion of sufficiency of hot leg injection for boiloff compensation was analyzed for the highest plausible system pressure to conservatively minimize the hot leg injection rate, which

decreases as system pressure increases. Thus, the system pressure was assumed to be at the main steam safety valve setting pressure of 8.273 MPa [1,200 psia] with the explanation that the hot leg switchover is expected to take place after one hour and the system pressure is thought to drop below the main steam safety valve setting pressure by that time.

Using the minimum hot leg injection flow rate of [
(Proprietary information withheld under 10 CFR 2.390)

] was assumed to envelop the maximum RWSP water transient temperature following a LOCA. The corresponding core decay heat power and time after trip were calculated for the other two criteria to determine that under the assumed conditions total hot leg injection flow is sufficient to flush the core at 5,600 seconds.

In conclusion, the applicant stated that the earliest time when the operator can switch over the injection lines to hot leg injection, considering the time restrictions derived from all three analyzed criteria, was 100 minutes (6,000 seconds). This earliest time when the operator can switch over the injection lines to hot leg injection bounds the switchover time restrictions based on the analyzed criteria by 400 sec [6.67 minutes]. To accept the response to RAI 719-5352, Question 15.6.5-88, and consider the RAI question resolved, the staff finds it necessary for the applicant to demonstrate the applicability of the Ishii-Grolmes liquid entrainment onset correlation with regard to the conditions under which it was applied in the analysis, as this correlation is not part of the MUAP-07013-P SBLOCA methodology. In addition, the applicant needs to consider the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79, and associated variation in the hot leg steam flow rate on the assessment of the earliest time when the operator can switch over to hot leg injection. Therefore, the staff tracks RAI 719-5352, Question 15.6.5-88 as an Open Item and proposes the following RAI for its resolution.

Open Item 15.6.5-13

RAI 861-6062, Question 15.6.5-97 (Follow-up to RAI 719-5352, Question 15.6.5-88):

> Demonstrate the applicability of the Ishii-Grolmes liquid entrainment onset correlation with regard to the conditions under which it was applied in the analysis provided in the response to RAI 719-5352, Question 15.6.5-88. Consider the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79, and associated variation in the hot leg steam flow rate on the assessment results for the earliest time when the operator can switch over to hot leg injection.

Fibrous debris, in combination with other types of debris, can bypass the US-APWR sump strainer and reach the reactor core region where fuel blockage can take place. Debris can cause fuel blockage near the reactor core inlet region in a direct vessel ECCS injection mode and, in a simultaneous ECCS injection mode, fuel blockage in the top core regions becomes possible. On March 17, 2011, the staff issued RAI 719-5352, Question 15.6.5-89, asking the applicant to demonstrate that fuel blockage at the core inlet will not preclude or adversely impact coolant mixing between the lower plenum and the core and show that fuel blockage by debris in the top core area will not interfere with downwards coolant penetration into the core region during the core flushing process. The applicant was also asked: (1) to discuss effects from fuel blockage by debris in the reactor coolant on the US-APWR boric acid precipitation evaluation, (2) if fluid mixing between the reactor lower plenum and adjacent core regions has been credited in the precipitation analysis, to demonstrate that fuel blockage at the core inlet will

not preclude or adversely impact coolant mixing between the lower plenum and the core, (3) and to show that fuel blockage by debris in the top core area will not interfere with downwards coolant penetration into the core region during the core flushing process.

On May 25, 2011, the applicant responded to RAI 719-5352, Question 15.6.5-89 [Reference 15-44], explaining that the possibility of the core inlet blockage due to debris is discussed in Section 4.4.1 (3), Revision 0 of MUAP-08013-P, “US-APWR Sump Strainer Downstream Effects,” December 2008 [Reference 15-45]. The applicant explained that the limiting scenario for boric acid precipitation is a cold leg break when the core inlet flow matches the core boiloff rate so that the core practically experiences stagnant flow conditions. In this case, the majority of the ECCS flow, injected via the DVI lines, spills through the cold leg break opening while the remaining small fraction of the safety injection liquid flows downward in the downcomer and into the core to make up for the core boil-off rate. In the RAI response, the applicant provided an assessment for the boric acid mixing flow rate between the core and the lower plenum regions to evaluate the fluid velocity at the core inlet that can be attributed only to the process of fluid mixing between the regions due to accumulation of boric acid in the core. In this analysis, the mixing flow was derived without considering the ECC water from the downcomer. Instead, it was assumed that heat transfer from the core to the downcomer liquid through the core barrel would heat the downcomer ECC water allowing it to flow directly into the core without effectively engaging the lower plenum volume. The assumption that 50% of the lower plenum volume participates in the boric acid mixing volume, introduced in the post-LOCA, long-term cooling evaluation of boric acid precipitation in US-APWR DCD Tier 2 Section 15.6.5.3.1.3, “Post-LOCA Long Term Cooling Evaluation Model,” and discussed in RAI 706-5339, Question 15.6.5-81, was used here to calculate the average boric acid concentration in the lower plenum. Thus, it was calculated as being half the average core boric acid concentration, which is provided in DCD Figure 15.6.5-42, ”US-APWR Post LOCA Long Term Cooling Evaluation for 14.7 psia.” Then, the boric acid influx rate for the lower plenum was calculated by differentiating the concentration curve for the region. Using a boric acid mass balance for the lower plenum, the mixing flow rate and the associated core inlet flow rate were calculated based on the lower plenum support plate flow area and the lower plenum water density at 3.9 C [39 °F] and atmospheric pressure. The calculated mixing flow rate and velocity, required to maintain the boric acid concentration in the lower plenum at half the value in the core region, were presented in Figure 3 of the RAI response. As seen from the figure, the predicted quantities are quite low. Based on this assessment outcome, the staff generally agrees with the conclusion by the applicant that the pressure loss will not be significantly increased if partial blockage occurs at the core inlet. However, in order to confirm that the impact of accumulated sump debris on the mixing process between the core and the lower plenum will remain very limited so that the assumptions underlying the boric acid precipitation analysis remain valid, the staff finds it necessarily to consider the following: the applicant is currently engaged in an ongoing experimental effort to assess the US-APWR core blockage by debris as part of MUAP-08013-P. Therefore, the staff defers the final acceptance of this part of the RAI response until MUAP-08013-P, “US-APWR Sump Strainer Downstream Effects,” becomes available and pertinent report findings are reviewed by the staff. Furthermore, with regard to sump debris effects during hot leg injection and consideration of eventual impacts on boric acid precipitation related to possible debris blockage interference with downwards coolant penetration into the core region during the core flushing process, the RAI response provided only a reference to Section 4.4.3 in Revision 0 of MUAP-08013-P, “US-APWR Sump Strainer Downstream Effects,” December 2008. Therefore, the staff tracks RAI 719-5352, Question 15.6.5-89 as an Open Item and proposes the following RAI for its resolution.

Open Item 15.6.5-14

RAI 861-6062, Question 15.6.5-100 (Follow-up to RAI 719-5352, Question 15.6.5-89):

> Provide an updated response to RAI 719-5352, Question 15.6.5-89 that takes into consideration relevant and conforming findings related to US-APWR core debris blockage that also accounts for any experimental test results to assess the US-APWR core blockage. Currently, such additional information is planned to be included in MUAP-080013-P, "US-APWR Sump Strainer Downstream Effects."

15.6.5.4.3.5 Conclusions

Pending resolution of open items, long-term cooling has not been assured.

15.6.5.4.4 List of Identified Open Items

This section lists the identified **OPEN ITEMS** as they pertain to large-break LOCA, small-break LOCA, and long-term cooling.

15.6.5.4.4.1 Large-Break LOCA Open Items

Open Item 15.6.5-1, RAI 352-2369, Question 15.6.5-3:

In RAI 15.6.5-3 and 15.6.5-4 the staff sought assurance that the HHSI flow curves were modeled in a conservative fashion. In its responses to RAI 15.6.5-3 and RAI 15.6.5-4 the applicant noted that HHSI flow characteristics for both minimum and maximum safeguards (Figures 6.3-15 and 6.3-16 of the DCD) are based on conservative assumptions and account for the head loss due to the accumulation of debris on the ECC/CS strainer. The head loss is discussed in "MUAP-08001(R5), "US-APWR Sump Strainer Performance," Revision 5.

The acceptance of the applicant's assumed ECC/CS strainer head loss has a direct bearing on the conservativeness of the ECC flow curve being used for the LOCA analyses. **An Open Item has been created in this SE as the staff has not yet completed the review of MUAP-08001.**

Open Item 15.6.5-2, RAI 352-2369, Question 15.6.5-15

RAI 15.6.5-15 asked if there had been any changes in the LBLOCA evaluation model since its review by the NRC. The applicant responded that there had been no changes. This response was appropriate at the time the response was given (July 2009). Subsequently, the LBLOCA analysis was revised. For Revision 2 of the DCD, the LBLOCA analysis was redone due to modifications to WCOBRA/TRAC and HOTSPOT. The modifications were evaluated by the staff as part of its SER for Topical Report MUAP-07011-P. For Revision 3 of the DCD, the LBLOCA analysis was redone again because of a re-analysis of the minimum containment pressure curve, modifications to WCOBRA/TRAC and HOTSPOT, changes in the uncertainty ranges for core power and SI fluid temperatures, and the addition of a bounding bias to accumulator flow. The staff's evaluation of the minimum containment pressure curve is contained in the SER for DCD Tier 2 subsection 6.2.1.5. The staff's evaluation of the uncertainty ranges for core power and SI fluid temperatures is given in the Topical Report

MUAP-07011-P SE [Reference 15-6]. **Open Item to verify that the accumulator flow rate bias used in the LBLOCA evaluation is conservative relative to that determined in the Advanced Accumulator Topical Report, MUAP-07001 (Reference 15-8). Also, an Open Item exists to track the closure of the minimum containment pressure curve is contained in the SER for DCD Tier 2 Subsection 6.2.1.5 and the SE on MUAP-07011-P (R3), "Large Break LOCA Code Applicability Report for US-APWR."**

15.6.5.4.4.2 Small-Break LOCA Open Items

Open Item 15.6.5-3, RAI 760-5576, Question 9.2.2-82:

Regarding Item II.K.3.25 on "Effect of Loss of AC Power on Pump Seals" the applicant responded that for the US-APWR, RCP seals were designed such that the pressure tightness (or leak tightness) is usually maintained by the No.1 seal. In case of a failure of the No.1 seal, the No. 2 seal can withstand full pressure as the defense-in-depth function. The applicant stated that RCP seal integrity is discussed in Chapter 8, Section 8.4.2.1.2 and Chapter 9, Section 9.2.2 of the DCD**. The ability of the No. 2 seal to withstand full pressure is still under staff review.**

Open Item 15.6.5-4

The completion of the accumulator uncertainty analysis review does not impact the review of the advanced accumulator model in M-RELAP5; however, these uncertainties need to be considered in the US-APWR SBLOCA safety analysis to determine the advanced accumulator flow rate. **Open Item: To verify that the accumulator flow rate bias used in MUAP-07025-P [Reference 47]is conservative relative to that determined in the Advanced Accumulator Topical Report, MUAP-07001 [Reference 15-8], which is still under staff review.**

15.6.5.4.4.3 Long-Term Cooling Open Items

Open Item 15.6.5-5, RAI 861-6062 Question 15.6.5-99

> US APWR DCD Tier 2 Section 15.6.5.3.1.3, "Post-LOCA Long Term Cooling Evaluation Model," states that the post-LOCA, long-term cooling evaluation model is similar to the model described in several references, including the following: "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P" and "Post LOCA Long Term Cooling Model" due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit, NRC letter dated November 23, 2005, D.S. Collins to G.C. Bischoff." Provide an explanation of how the subject document relates to the US APWR post-LOCA, long-term cooling evaluation model and how any non-conservatisms are treated.

Open Item 15.6.5-6, RAI 861-6062, Question 15.6.5-95

> In the response to RAI 706-5339, Question 15.6.5-79, MHI stated that the axial power shape affects the core average void fraction. MHI then performed a sensitivity calculation where the core average void fraction was reduced to 80 percent of the base case value. Provide a basis for the 80 percent value.

Using a uniform axial power shape and reducing the core average void fraction may not be conservative relative to using different axial power shapes. For example, a top peaked power shape may result in the same core average void fraction and result in the top of the core being exposed. Provide the results of a thermal-hydraulic analysis quantifying the two-phase mixture level in the US-APWR reactor during the long-term cooling phase assuming the most limiting break size, break location, and ECCS performance conditions. The analysis should include loop seal piping becoming plugged as well as the most limiting axial power shape.

In the axial power sensitivity, the reactor vessel pressure differential falls to a value just below the loop pressure differential for a short time. This implies that the core mixture level is below the bottom of the hot leg elevation. At the same time, the conclusion is that the two-phase mixture level is always maintained above the bottom of the hot leg elevation. How can the conclusion statement be made when the condition that is observed does not meet the requirement?

Open Item 15.6.5-7, RAI 861-6062, Question 15.6.5-96

Further information is needed on the Yeh correlation. The response to RAI 706-5339, Question 15.6.5-80, includes Figure 1, which compares predicted versus measured void fractions including test data at low pressure (20 psia [0.14 MPa]). However, it is not clear how many tests were actually run at this low pressure. Provide a figure showing clearly the comparison against low pressure test data and include a table that lists the test flow conditions and measured void for each data point used in assessing the correlation at low pressure.

Open Item 15.5.6-8, RAI 719-5352, Question 15.6.5-90

The response to RAI 719-5352, Question 15.6.5-90, refers to Revision 0 of MUAP-07019-P, "Qualification of Nuclear Design Methodology using PARAGON/ANC," December 2007. Therefore, the closure of this RAI question is conditional on the NRC's approval of MUAP-07019-P Revision 0. Provide the status of NRC's approval of MUAP-07019-P Revision 0.

Open Item 15.6.5-9, RAI 861-6062, Question 15.6.5-98

1. In Table RAI-15.06.05-90.1 of UAP-HF-11106, the bounding boron concentration for the lower plenum coolant prior to deborated condensate water entering the lower plenum is given as [(Proprietary information withheld under 10 CFR 2.390)

   ]
   identified as the maximum deborated water volume cases, demonstrate that the entire RWSP volume will be injected into the RCS prior to re-establishment of natural circulation conditions. Provide total water mass (liquid and vapor) out the break, total water mass and volume injected from the RWSP and accumulators between the start of reflux condensation to resumption of natural circulation. If the entire RWSP volume of water is not capable of being injected and mixed with the initial RCS boron concentration how is the RPV boric acid concentration determined at the time of natural recirculation resumption?

2. Figures RAI 15.06.05-85.5 and RAI 15.06.05-85.6 in the response to RAI 718-5402, Question 15.6.5-85, provide the amount of condensate generated in the SG U-tubes [(Proprietary information withheld under 10 CFR 2.390)]. The amount of condensate generated by reflux condensation depends on the break size and the number of SGs receiving emergency feedwater and can be considerably larger than the volume of the loop seals. Assess the impact and provide the analysis results that account for generation and possible RCS condensate accumulation [(Proprietary information withheld under 10 CFR 2.390)] as considered in the recriticality consequences evaluation in the responses to RAI 719-5352, Questions 15.6.5-90 and 15.6.5-91?

3. The initial RCS coolant boron concentration was determined at [(Proprietary information withheld under 10 CFR 2.390)

   ] If not, why is one not needed?

4. In Table RAI-15.06.05-90.1 of UAP-HF-11106 the initial HFP, ARO boron concentration of [(Proprietary information withheld under 10 CFR 2.390)

   ] is the minimum margin condition? If not please justify.

5. What is the basis for the applied formula for calculating the boron concentration of coolant entering the core following deborated slug insertion that is provided in Table RAI-15.06.05-90.1 in the response to RAI 719-5352, Question 15.6.5-90? Is there experimental evidence that supports the appropriateness of this simplistic equation for the purposes of the analysis? Compare the formula predictions against test data and demonstrate that the formula, if and as applied in the US-APWR boron dilution analysis, leads to conservative results.

Open Item 15.6.5-10, RAI 861-6062, Question 15.6.5-92

US-APWR DCD Tier 2 Subsection 15.6.5.3.1.3 “Post-LOCA Long Term Cooling Evaluation Model” describes the model that is used to predict the boric acid concentration in the reactor core during the post-LOCA, long-term cooling period. A more detailed description of the model is provided in Appendix B to UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1” as part of the response to RAI Question 15.6.5-44. Identify the decay heat model used in the US-APWR boron precipitation analyses performed with the long-term cooling evaluation model. Provide the decay heat multiplier assumed in the calculations. The amount of liquid in the mixing volume depends on the predicted vapor volumetric fraction within this volume. The applied axial power profile can impact the volumetric vapor fraction in the core region. If the US-APWR boron precipitation analyses were not produced with a limiting bottom-peaked axial power profile, provide the impact on the precipitation timing

resulting from an assumed limiting bottom-peaked axial power shape. Also discuss the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79.

Open Item 15.6.5-11, RAI 861-6062, Question 15.6.5-93

According to Appendix B, "Post-LOCA Long Term Cooling Evaluation Model for US-APWR," to the RAI 352-2369, Question 15.6.5-44, response, the boric acid precipitation calculation is initiated at the beginning of the reflood phase to determine the timing of boric acid precipitation. However, the response to RAI 352-2369, Question 15.6.5-45, refers to the point in time when the operators recognize a LOCA to characterize the timing of manual switchover to hot leg injection and states that the related operator action procedures will be provided in future Emergency Procedure Guidelines (EPGs). Explain how it will be ensured in the EPGs that the timing of such manual switchover to hot leg injection will be defined so that it occurs prior to the point of boric acid precipitation as predicted by post-LOCA, long-term cooling evaluation model.

Open Item 15.6.5-12, RAI 861-6062, Question 15.6.5-94

In response to RAI 352-2369, Question 15.6.5-48, it was stated that the mixing volume makeup flow rate is the sum of core evaporation rate and the change rate of the mixing volume water mass. Compare the calculated makeup flow rate using this method with the flow rate that would be obtained from the pressure difference between the downcomer and core. Also compare this flow rate to the total HHIS flow rate. It was also stated in the original RAI response that "In the post-LOCA long-term cooling period, downcomer water level is maintained more than cold leg bottom elevation by HHIS." Explain how the downcomer water level can be above the cold leg bottom for a large cold leg break (double ended guillotine break).

Open Item 15.6.5-13, RAI 861-6062, Question 15.6.5-97

Demonstrate the applicability of the Ishii-Grolmes liquid entrainment onset correlation with regard to the conditions under which it was applied in the analysis provided in the response to RAI 719-5352, Question 15.6.5-88. Consider the impact of possible loop seal plugging, discussed in RAI 706-5339, Question 15.6.5-79, and associated variation in the hot leg steam flow rate on the assessment results for of the earliest time when the operator can switch over to hot leg injection.

Open Item 15.6.5-14, RAI 861-6062, Question 15.6.5-100

Provide an updated response to RAI 719-5352, Question 15.6.5-89 that takes into consideration relevant and conforming findings related to US-APWR core debris blockage that also accounts for any experimental test results to assess the US-APWR core blockage. Currently, such additional information is planned to be included in Revision 2 of MUAP-080013-P, "US-APWR Sump Strainer Downstream Effects," scheduled for release by MHI on August 31, 2011.

15.6.5.5 Combined License Information Items

There are no COL information items from Table 1.8-2 of the DCD pertaining to this section.

15.6.5.6 Conclusions

As a result of the open items, the staff is unable to finalize its conclusion on Section 15.6.5 in accordance with the requirements of 10 CFR 50.46, GDC 13, GDC 35, 10 CFR 100 and 10 CFR 50.67.

15.6.5.7 References

15-1 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, Reference UAP-HF-08153, "Subject: Submittal of US-APWR Design Control Document Revision 1 in Support of Mitsubishi Heavy Industries, Ltd.'s Application for Design Certification of the US-APWR Standard Plant Design," dated August 29, 2008. (ADAMS Accession ML082480524).

15-2 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, Reference UAP-HF-09490, "Subject: Submittal of US-APWR Design Control Document Revision 2 in Support of Mitsubishi Heavy Industries, Ltd.'s Application for Design Certification of the US-APWR Standard Plant Design," dated October 27, 2009 [ADAMS Accession No. ML093070344]. Supplemented by Letter from Yoshiki Ogata General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, Reference UAP-HF-10290 "Subject: Transmittal of US-APWR DCD Chapter 15.6.5 SBLOCA Markups," dated November 1, 2010 [ADAMS Accession No. ML103120093].

15-3 Code of Federal Regulation, Title 10, "Energy," Part 50 - Domestic Licensing of Production and Utilization Facilities, §50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

15-4 Code of Federal Regulation, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix K, "ECCS Evaluation Models."

15-5 Mitsubishi Heavy Industries, Ltd., "Large Break LOCA Code Applicability Report for US-APWR", MUAP-07011-P (R1), March 31, 2011 (ML111101049).

15-6 Safety Evaluation Report on MUAP-07011-P (R3), "Large Break LOCA Code Applicability Report for US-APWR," April 2012 (Draft).

15-7 Mitsubishi Heavy Industries letter UAP-HF-09384, "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1," dated July 2009 (ADAMS Accession Numbers: cover letter: ML092040050; Enclosure 2: ML092040051).

15-8 Mitsubishi Heavy Industries, Ltd., letter UAP-HF-11283, "Revision 4 of the Topical Report MUAP-07001-P 'The Advanced Accumulator,'" MUAP-07001-P (R4), August 31, 2011.

15-9 Mitsubishi Heavy Industries, Ltd., “Transmittal of Revision 1 to Topical Report MUAP-07013 ‘Small Break LOCA Methodology for US-APWR,’” UAP-HF-10145, May 31, 2010 (ADAMS Accession ML101610401 and ML101610402).

15-10 RELAP5-3D Code Manual Volume I: “Code Structure, System Models, and Solution Methods,” INEEL-EXT-98-00834, Revision 2.4, June 2005.

15-11 USNRC, “Safety Evaluation Report for Mitsubishi Heavy Industries Small Break LOCA Methodology for US-APWR, MUAP-07013-P, Revision 2,” December, 2010 (ADAMS Accession No. ML112070775).

15-12 Letter from Masahiko Kaneda, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. R. William Borchardt Director, Office of New Reactors, Project No.0751 MHI Ref: UAF-HF-07185, “Subject: Technical Report on Small Break LOCA Sensitivity Analyses for US-APWR (MUAP-07025) Submitted in Support of US-APWR Design Certification Application,” dated December 31, 2007 (ADAMS Accession ML080370082).

15-13 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, MHI Ref: UAF-HF-10146, Subject: Transmittal of Revision 1 to Technical Report MUAP-07025 “Small Break LOCA Sensitivity Analyses for US-APWR,” dated May 31, 2010. (ADAMS Accession No. ML101580036).

15-14 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, MHI Ref: UAF-HF-10288, Subject: Transmittal of Revision 2 to Technical Report MUAP-07025 "Small Break LOCA Sensitivity Analyses for US-APWR,” dated November 1, 2010. (ADAMS Accession No. ML103120119).

15-15 Not used
15-16 Not used
15-17 Not used

15-18 Letter from Mike Takacs, Project Manager, US-APWR Projects Branch, Division of New Reactor Licensing, Office of New Reactors, USNRC to Mr. Yoshiki Ogata, General Manager, APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., 16-5, Konan 2-Chome, Minato-Ku, Tokyo, 108-8215 Japan, Subject: Revised Public RAIs 3rd set for SBLOCA Topical Report MUAP-07013, dated September 10, 2009 (ADAMS Accession ML092600254).

15-19 Letter from Mike Takacs, Project Manager, US-APWR Projects Branch, Division of New Reactor Licensing, Office of New Reactors, US NRC to Mr. Yoshiki Ogata, General Manager, APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., 16-5, Konan 2-Chome, Minato-Ku, Tokyo, 108-8215 Japan, Subject: 5th set RAIs SBLOCA Topical Report MUAP-07013, dated February 17, 2010 (ADAMS Accession ML100550571).

15-20 R. Beaton and D. Fletcher, “US-APWR SBLOCA RELAP5/MOD3.3 Confirmatory Runs,” Information Systems Laboratories, Inc., ISL-NSAO-TR-09-09, Revision 1 (Proprietary) August 2009. (ADAMS Accession MLXXXXXXXXXX).

15-21 Mitsubishi Heavy Industries letter UAP-HF-10038, “MHI's Response to US-APWR DCD RAI No. 513-4170 Revision 0,” dated February 2010 (ADAMS Accession Nos.: cover letter: ML100480087; Enclosure 2: ML100480088).

15-22 Mitsubishi Heavy Industries letter UAP-HF-10039, “MHI's 1st Response to US-APWR DCD RAI No. 514-4040 Revision 2,” dated February 2010 (ADAMS Accession Nos.: cover letter: ML100480206; Enclosure 2: ML100480207).

15-23 Mitsubishi Heavy Industries letter UAP-HF-10042, MHI's 2nd Response to US-APWR DCD RAI No. 514-4040 Revision 2, dated February 2010 (ADAMS Accession Nos.: cover letter: ML100500491; Enclosure 2: ML100500492).

15-24 Mitsubishi Heavy Industries letter UAP-HF-09492, MHI's 1st Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0), "Small Break LOCA Methodology for US-APWR" on September 8, 2009, dated October 23, 2009 (ADAMS Accession No. ML093000526).

15-25 Mitsubishi Heavy Industries letter UAP-HF-09512, MHI's 2nd Response to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR," on 09/08/2009, dated November 11, 2009. (ADAMS Accession No. ML093160363).

15-26 Mitsubishi Heavy Industries letter UAP-HF-10040, Supplementary Response to RAI CA-5 on M-RELAP5 Topical Report MUAP-07013-P (RO), “M-RELAP5 Code Modification for Break Flow Noding Sensitivity Calculations” on February 10, 2010 (ADAMS Accession No.: ML100541342).

15-27 Mitsubishi Heavy Industries letter UAP-HF-10059, MHI’s 1st Response to NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) “Small Break LOCA Methodology for US-APWR” on February 16, 2010, March 2010 (ADAMS Accession Nos.: cover letter: ML100690153; Enclosure 2: ML100690154).

15-28 Mitsubishi Heavy Industries letter UAP-HF-10348, Revision to MHI's RAI Responses on US-APWR DCD Chapter 15.6.5 SBLOCA on December 28, 2010 (ADAMS Accession Nos.: cover letter: ML110100359; Enclosure 2: ML110100360).

15-29 R. R. Schultz, "RELAP5-3D© Code Manual Volume V: User’s Guidelines," INEEL-EXT-98-00834, Volume V, Revision 2.4, June 2005.

15-30 NUREG/CR-5535, "RELAP5/MOD3.3 Code Manual Volume II: User's Guide and Input Requirements," Patch03 Version, 2007.

15-31 R. Beaton and D. Fletcher, Updated US-APWR SBLOCA RELAP5/MOD3.3 Confirmatory Runs, ISL-NSAO-TR-10-16, dated December, 2010 (ADAMS Accession MLXXXXXXXXXX).

15-32 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, Reference: UAP-HF-10287, Subject: Transmittal of Revision 2 to Topical Report MUAP-07013 'Small Break LOCA Methodology for US-APWR, dated November 1, 2010 (ADAMS Accession No. ML103120120 and ML103120208).

15-33 Letter from Masahiko Kaneda, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, LTD., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. David B. Matthews Project No.0751, MHI Ref: UAP-HF-07092,"Subject: Transmittal of the Topical Report entitled "Small Break LOCA Methodology for US-APWR," July 20, 2007 (ADAMS Accession No. ML072150095).

15-34 Mitsubishi Heavy Industries letter UAP-HF-09041, MHI's 2ND Part Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on December 5, 2008, dated February 13, 2009 (ADAMS Accession No. ML090490844).

15-35 Request for Additional Information 706-5339 Revision 0, dated March 1, 2011(ADAMS Accession No. to be provided).

15-36 UAP-HF-11130, Docket No. 52-021, "MHI's Response to US-APWR DCD RAI No. 706-5339 Revision 0," April 2011.

15-37 UAP-HF-11083, Docket No. 52-021, "MHI's Response to US-APWR DCD RAI No. 706-5339 Revision 0," March 2011.

15-38 Request for Additional Information 718-5402 Revision 0, dated March 17, 2011. (ADAMS Accession No. to be provided).

15-39 Request for Additional Information 719-5352 Revision 0, dated March 17, 2011 (ADAMS Accession No. to be provided).

15-40 UAP-HF-11136, Docket No. 52-021, "MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0," May 2011.

15-41 UAP-HF-11106, Docket No. 52-021, "Enclosure 2, 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0," May 2011.

15-42 UAP-HF-11138, Docket No. 52-021, "Enclosure 2, 2nd MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0," May 2011.

15-43 UAP-HF-11130, Docket No. 52-021, "Enclosure 2, MHI's Response to US-APWR DCD RAI No. 719-5339 Revision 0," April 2011.

15-44 UAP-HF-11139, Docket No. 52-021, "3rd MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0," May 2011.

15-45 Mitsubishi Heavy Industries, Ltd., "US-APWR Sump Strainer Downstream Effects," MUAP-08013-P, Revision 0, December 2008.

15-46 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, Reference UAP-HF-11078, "Subject: Submittal of US-APWR Design Control Document Revision 3 in Support of Mitsubishi Heavy Industries, Ltd.'s Application for Design Certification of the US-APWR Standard Plant Design," dated March 31, 2011 [ADAMS Accession No. ML110980238].

15-47 Letter from Yoshiki Ogata, General Manager- APWR Promoting Department, Mitsubishi Heavy Industries, Ltd., to Document Control Desk, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, Attention: Mr. Jeffrey A. Ciocco, MHI Ref: UAF-HF-11080, Subject: Transmittal of Revision 3 to Technical Report MUAP-07025 "Small Break LOCA Sensitivity Analyses for US-APWR," dated March 31, 2011 (ADAMS Accession No. ML110960348).

15-48 Mitsubishi Heavy Industries letter UAP-HF-09417, MHI's 2ND Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 06/11/2009, dated August 11, 2009. (ADAMS Accession Nos. ML090490844 and ML092300135).

15.8 Anticipated Transients Without Scram

15.8.1 Introduction

An anticipated transient without scram (ATWS) is an AOO followed by failure of the RT portion of the protection system.

15.8.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 15.8, summarized here as follows:

MHI states that the design features of the US-APWR include a diverse actuation system (DAS), which is described in Section 7.8.3 of the DCD. In accordance with the ATWS Rule (10 CFR 50.62) the DAS is diverse from the existing RTS from sensor output to the final actuation device. The DAS automatically initiates the EFW system and a turbine trip under conditions indicative of an ATWS. The DAS also includes a diverse means of interrupting power to the reactor trip breakers in the event the ATWS is caused by a common cause failure of the reactor trip. Chapter 19 demonstrates that the contribution of the ATWS to the total core damage frequency meets the safety goal of less than 10^{-5} per reactor year.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.8.3 Regulatory Basis

The relevant requirements, and the associated acceptance criteria, are given in Section 15.8 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 15.8 of NUREG-0800.

The relevant requirements are:

1. 10 CFR 50.62 (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via: (a) inclusion of prescribed design features, and (b) demonstration of their adequacy.

2. 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry.

3. GDC 12, as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed.

4. GDC 14, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary.

5. GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of PAs.

6. GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reactions must be limited to negligible amounts.

7. GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment.

8. GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

Acceptance criteria adequate to meet the above requirements include:

1. Provide measures to automatically initiate the EFW system and a turbine trip under conditions indicative of an ATWS. This equipment shall be independent and diverse from the reactor trip system from sensor output to the final actuation device.

2. Either provide a diverse scram system satisfying the design and quality assurance criteria specified in SRP Section 7.2 or demonstrate that the consequences of an ATWS event are within acceptable values.

3. These system and equipment shall be demonstrated to provide reasonable assurance that unacceptable plant conditions do not occur in the event of an ATWS.

4. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either the criteria are met or a diverse scram system is installed that reduces significantly the probability of a failure to scram.

15.8.4 Technical Evaluation

The first acceptance criterion, to automatically initiate the EFW system and a turbine trip under conditions indicative of an ATWS with independent and diverse equipment, is part of the ATWS Rule (10 CFR 50.62). MHI describes how the US-APWR DAS conforms to the ATWS Rule (including this criterion) in DCD Section 7.8.3, “Diverse Instrumentation and Control Systems” and in Appendix B to MUAP-07006-P-A “Defense in Depth and Diversity.” As stated in Section 7.8.3 of this SER, the staff finds the US-APWR DAS design satisfies the specific design requirements identified in the ATWS Rule. As discussed in DCD Section 7.8.3, the US-APWR DAS design also meets the second criterion because it includes a scram system that is independent and diverse from the RTS from sensor output to the points of interruption of power to the control rods.

The technical report MUAP-07014-P, “Defense-In-Depth and Diversity Coping Analysis” contains an analysis for each Chapter 15 event assuming all of the safety functions of the digital control system are disabled by a common cause failure (CCF). Because a digital safety system CCF is a beyond design basis event, the coping analysis takes credit for operation of the non-safety DAS. The results of the AOO events in MUAP-07014-P bound an ATWS analysis because both RTS and ESF are assumed to fail. The staff review of MUAP-07014-P is ongoing and will be tracked with **Open Item 15.08-1.**

Chapter 19 demonstrates that the contribution of ATWS to the total core damage frequency is below the SECY-83-293, “Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events,” goal of 10^{-5} per reactor year.

15.8.5 Combined License Information Items

There are no COL information items from DCD Tier 2, Table 1.8-2 that affect this section.

15.8.6 Conclusions

As a result of the open item, the staff is unable to finalize its conclusion on Section 15.8 in accordance with the requirements of 10 CFR 50.62.