

Response to

Request for Additional Information No. 34, Revision 0

8/01/2008

U. S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 15 - Introduction - Transient and Accident Analyses

SRP Section: 15.01.01 - 15.01.04 - Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

SRP Section: 15.01.05 - Steam System Piping Failures Inside and Outside of Containment (PWR)

SRP Section: 15.02.01-15.02.05 - Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)

SRP Section: 15.02.06 - Loss of Non-Emergency AC Power to the Station Auxiliaries

SRP Section: 15.02.07 - Loss of Normal Feedwater Flow

SRP Section: 15.02.08 - Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

SRP Section: 15.03.03-15.03.04 - Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

SRP Section: 15.05.01-15.05.02 - Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

SRP Section: 15.06.01 - Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve

SRP Section: 15.06.03 - Radiological Consequences of Steam Generator Tube Failure (PWR) 07/1981

Application Section: FSAR Ch. 15.0

SRSB Branch

Question 15-1:

Please provide the assessment for the parameters, initial conditions, and single failures used in various transients and accidents to support the values and sequence of events assumed in each event that would lead to the most conservative results with respect to each acceptance criteria.

Response to Question 15-1:

A response to this question will be provided by October 30, 2008.

Question 15-2:

Confirm that each of the transient and accident analyzed has been assessed against multiple acceptance criteria including specified acceptable fuel design criteria (SAFDL), the maximum primary pressure, maximum secondary side pressure, and the minimum departure from nucleate boiling ratio (MDNBR).

Response to Question 15-2:

A response to this question will be provided by October 30, 2008.

Question 15-3:

Please extend the tables shown in chapter 15.0 to show for each transient and accident the limiting power, temperatures, flows, levels, scram reactivity, reactivity coefficients, heat transfer coefficients, and degree of SG tube plugging.

Response to Question 15-3:

A response to this question will be provided by October 30, 2008.

Question 15-4:

Please provide in the analysis section of each transient and accident the technical bases to demonstrate that the limiting single failure is selected. Specifically state the physical bases. Consider each acceptance criteria and show that the limiting single failure is selected for each acceptance criteria (for example minimum DNBR, peak RCS pressure, and peak secondary side pressure may differ with the single failure assumed in the analysis.)

In addition to providing an explanation in each section of the SAR please include the information in a summary table in section 15.0 of the SAR.

Response to Question 15-4:

A response to this question will be provided by October 30, 2008.

Question 15-5:

Please provide in the analysis section of each transient and accident the technical bases to demonstrate that the limiting analysis conditions are selected. Specifically state the physical or phenomenological bases with respect to 1) why retention or loss of offsite power is limiting, 2) why beginning or end of cycle is limiting, 3) why HFP or HZP or an intermediate power is limiting. Consider each acceptance criteria and show that limiting conditions are selected for consideration of each criteria (for example minimum DNBR, peak RCS pressure, and peak secondary side pressure may differ with limiting analysis conditions.)

In addition to providing an explanation in each section of the SAR include the information in a summary table in section 15.0 of the SAR.

Regulatory basis: SRP acceptance criteria instructs the reviewer to verify “*whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.*”

RG 1.206, C.I.15.6.2 *Sequence of Events and Systems Operation* states
“*The applicant should discuss the following considerations for each initiating event:...*”

Response to Question 15-5:

A response to this question will be provided by October 30, 2008.

Question 15-6:

For each transient and accident describe the specific S-RELAP model if it is different from the generic base model. It appears that some transients have been evaluated using a single channel core while other transients have used a multiple core channel model. Please provide directly or by reference a description of the S-RELAP 5 model including nodalization used in non-LOCA asymmetric calculations.

Response to Question 15-6:

A response to this question will be provided by October 30, 2008.

Question 15-7:

In computing the reactor power for the transients analyzed in Sections 15.1 through 15-6, provide the decay heat curve used to calculate the total core thermal power. Specify the model that was used to calculate the decay power curve along with any assumptions made. Use a logarithmic scale over an appropriate range to display this quantity

(SRP 15.2.1-15.2.5 Section III.6.A, 15.2.6 Section III.5.A, 15.2.7-15.2.8 Section III)

Response to Question 15-7:

A response to this question will be provided by October 30, 2008.

Question 15-8:

Several of the transients analyzed in this chapter assumed the reactor tripped on a DNBR trip. Please explain how the system code interacts with the computer code which calculates the DNBR trip.

Response to Question 15-8:

A response to this question will be provided by October 30, 2008.

Question 15.01.01 - 15.01.04-1:

Provide the representative DNBR as a function of time for all Chapter 15.1 analyses for which it is a key parameter.

Regulatory basis: SRP 15.0, p. 15.0-10. *“The reviewer ensures that the applicant has presented the results of the analyses, including key parameters as a function of time...”*

...List all single failures or operator errors considered in the transient and accident analysis, and identify the limiting single failure for each event

Response to Question 15.01.01 - 15.01.04-1:

A response to this question will be provided by October 30, 2008.

Question 15.01.01 - 15.01.04-2:

Several tables in section 15.1 show a scram worth of -7353 pcm/F. Please clarify the units being used for scram worth.

Regulatory basis: SRP 15.0, p. 15.0-7. *"The reviewer ensures that the application contains the key plant parameters...."*

Response to Question 15.01.01 - 15.01.04-2:

See the response to Question 15.01.01 - 15.01.04-7.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 15.1-1, Table 15.1-4, Table 15.1-7, and Table 15.1-10 will be revised as described in the response to Question 15.01.01 - 15.01.04-7 and indicated on the enclosed markup.

Question 15.01.01 - 15.01.04-3:

In Section 15.1.1, an instantaneous reduction in feedwater temperature of 100 F is assumed in the analysis. Provide the quantitative assessment showing that this conservatively bounds the possible physical reduction in feedwater temperature.

Regulatory basis: SRP 15.0, p. 15.0-8. *“For each initiating evaluated, the reviewer ensures that the application includes a description of the occurrences that can lead to the event”*

Response to Question 15.01.01 - 15.01.04-3:

A typical winter temperature drop due to the loss of the feedwater heater train at hot full power (HFP) is 65°F (50°F plus 15°F uncertainty). However, an instantaneous drop of 100°F is assumed for this event for additional conservatism. The lower feedwater temperature increases the impact of the overcooling and a 35°F extra margin should bound any heater supplier and plant variation that may arise with the detailed plant design.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.01 - 15.01.04-4:

This event addressed in Section 15.1.1 assumes the minimum RCS flow allowed by plant TSs. Explain why this assumption is conservative.

Regulatory basis: SRP 15.1.1-15.1.4, p. 5. "*The values of the parameters used in the analytical model should be suitably conservative.*"

Response to Question 15.01.01 - 15.01.04-4:

The assumed Technical Specification minimum reactor coolant system (RCS) flow rate is conservative because it results in a lower minimum departure from nucleate boiling ratio (DNBR). In addition, the lower flows result in a larger temperature difference across the core and steam generators. This causes a decrease in the cold leg coolant temperatures that, in combination with a negative moderator temperature coefficient, maximizes the reactor power increase. This further increases the impact of the overcooling.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.01 - 15.01.04-5:

Figure 15.1-7 shows the SG pressure exceeding the MSRIV setpoint of 1384 psia. Explain why the MSRTs did not actuate. If the MSRTs did actuate it would result in all 4 SGs blowing down until the MSRCVs close. Please present the analysis results of such a scenario.

Regulatory basis: SRP 15.1.1-15.1.4, p. 3. *"The reviewer reviews the values ...for conformance to plant design."*

Response to Question 15.01.01 - 15.01.04-5:

A response to this question will be provided by October 30, 2008.

Question 15.01.01 - 15.01.04-6:

Please explain the apparent asymmetric temperatures shown in Fig 15.1-4

Response to Question 15.01.01 - 15.01.04-6:

The asymmetric cold leg response of Loop 3 shown in U.S. EPR FSAR, Tier 2, Figure 15.1-4—Decrease in Feedwater Temperature - Cold Leg Temperatures is a result of the pressurizer being located on this loop and the loss of mixing after the reactor coolant pumps (RCP) trip. As the reactor coolant system (RCS) volume shrinks due to the temperature decrease resulting from the decrease in feedwater temperature, there is an out-surge of higher temperature water from the pressurizer between 20 seconds and 60 seconds. The resulting temperature increase in the Loop 3 hot leg is sufficient to keep the cold leg temperature exiting the Loop 3 steam generator (SG) elevated above the other three loops.

The heat transfer rate across all four steam generators is fairly consistent (as shown in U.S. EPR FSAR, Tier 2, Figure 15.1-3—Decrease in Feedwater Temperature - Heat Transfer to Steam Generators) with SG 3 having a slightly higher heat transfer between 30 and 60 seconds. At approximately 60 seconds, the cold leg temperature reaches a minimum (as shown in U.S. EPR FSAR, Tier 2, Figure 15.1-4) in all four loops and (as shown in U.S. EPR FSAR, Tier 2, Figure 15.1-8—Decrease in Feedwater Temperature - Pressurizer Level), the pressurizer level reaches a minimum at approximately the same time.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.01 - 15.01.04-7:

Please explain the rod worth shown in figure 15.1-6. Show how the worth in \$ equals the worth in pcm divided by beta.

Response to Question 15.01.01 - 15.01.04-7:

The decrease in feedwater temperature event (U.S. EPR FSAR, Tier 2, Section 15.1.1) is performed at hot full power and end of cycle conditions. The scram worth given in U.S. EPR FSAR, Tier 2, Table 15.1-1—Decrease in Feedwater Temperature - Key Input Parameters, Table 15.1-4—Increase in Feedwater Flow - Key Input Parameters, Table 15.1-7—Increase in Steam Flow - Key Input Parameters, and Table 15.1-10—Inadvertent Opening of an SG Relief or Safety Valve – Key Input Parameters should be -7353 pcm (percent millirho) and not “pcm/°F.” The value of beta at this core condition is 0.005151. The total scram worth is calculated as:

$$R_{scram} = -\frac{(7353 \text{ pcm})}{\left(0.005151 \frac{\Delta\rho}{\$}\right)\left(10^5 \frac{\text{pcm}}{\Delta\rho}\right)} = -\$14.27 \quad [\text{Eq. 1}]$$

This is the value presented in U.S. EPR FSAR, Tier 2, Figure 15.1-6—Decrease in Feedwater Temperature – Reactivity for scram worth.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 15.1-1, Table 15.1-4, Table 15.1-7, and Table 15.1-10 will be revised as described in the response and indicated on the enclosed markup.

Question 15.01.01 - 15.01.04-8:

The asymmetry of this event addressed in Section 15.1.2 is increased if an increase in feedwater flow to one generator is accompanied by a decrease in feedwater flow to the other steam generators. Please explain the basis for the 150% increase in feedwater flow to one steam generator with no decrease in feedwater flow to the remaining SGs.

Regulatory basis: SRP 15.1.1-15.1.4, p. 5. *"The values of the parameters used in the analytical model should be suitably conservative."*

Response to Question 15.01.01 - 15.01.04-8:

A response to this question will be provided by September 24, 2008.

Question 15.01.01 - 15.01.04-9:

Figure 15.1-18 shows the SG pressure exceeding the MSRIV setpoint of 1384 psia. Explain why the MSRTs did not actuate. If the MSRTs did actuate it would result in all 4 SGs blowing down until the MSRCVs close. Please present the analysis results of such a scenario.

Regulatory basis: SRP 15.1.1-15.1.4, p. 3. *"The reviewer reviews the values ...for conformance to plant design."*

Response to Question 15.01.01 - 15.01.04-9:

A response to this question will be provided by October 30, 2008.

Question 15.01.01 - 15.01.04-10:

It appears that a sectorized core model was used for the analysis addressed in Section 15.1.2. In EMF-2310 the sectorized core model was applied only to the main steam line break event. The use of the model for the increase in feedwater flow event needs to be discussed fully. In particular, explain how the power for each core region is calculated.

Regulatory basis: SRP 15.1.1-15.1.4, p. 3. *“The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff”*

The SER for EMF-2320 states *“The staff also notes, however, that a generic topical report describing a code such as S-RELAP5 cannot provide full justification for each specific individual plant application. The individual applicant must still provide justification for each specific application of the code which is expected to include as a minimum, the nodalization...”*

Response to Question 15.01.01 - 15.01.04-10:

The increase in feedwater flow is a single steam generator cooldown event that results in asymmetric conditions within the reactor core. Departure from nucleate boiling ratio (DNBR) analysis of these asymmetric conditions using a single channel core model extends beyond a single loop transit time (10.5 seconds). To provide a more realistic reactivity feedback model, a split (or sectorized) core model is used for this event.

The split (or sectorized) channel core model was developed by modifying the non-LOCA single channel core S-RELAP5 model to partition its single-channel core (and the other flow paths within the reactor vessel) into a two-channel arrangement, with no crossflow or mixing between the affected region (representing $\frac{1}{4}$ of the core/reactor vessel cross-section) and the unaffected region (representing the remaining $\frac{3}{4}$ of the core/reactor vessel cross-section). This split core/reactor vessel arrangement is very similar to that used in analyzing the EMF-2310 methodology for the post-scrum main steam line break (MSLB) event with the following exceptions:

1. The active core region below the stuck-out rod cluster control assembly (the stuck rod region) is lumped together with the affected core region.
2. The reactor vessel upper head is not subdivided into three stacked sub-volumes because upper head voiding is not expected.
3. The upper head temperature initialization flow path of the single-channel core model is retained, but is split into affected and unaffected regions.
4. The upper plenum nodalization scheme of the single-channel core model is retained, but is split into affected and unaffected regions. Loop 4 is connected to the affected region.

The power for each core region is calculated by modeling the core fuel heat structures from the MSLB model to reflect the partitioning of core volume modifications with the following exceptions:

1. The fuel rods of the stuck-out-rod cluster control assembly (RCCA) core region are grouped together with the affected core region fuel rods to create a combined affected core region heat structure.
2. The fuel heat structures are changed from those calculated by the MSLB reactivity/power control system to those that are internally calculated from the reactor power and multipliers that correspond to the axial power shape of the single-channel core model, which are weighted by the number of fuel assemblies in the two core regions (namely, 60/241 weighting for the affected region, and 181/241 for the unaffected region).

To provide a more realistic moderator and Doppler reactivity feedback model, the active core coolant volume weighting factors are modified to reflect the partitioning of the single channel core into a split core, as described above. This is also done for the fuel heat structure/Doppler weighting factors.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.01 - 15.01.04-11:

Clarify Table 15.1-11 as to which MSRCV (SG-3 or SG-4) is failed open.

Regulatory basis: SRP 15.1.1-15.1.4, p. 2. "*The topics covered in the review include: ...valve malfunctions.*"

Response to Question 15.01.01 - 15.01.04-11:

A response to this question will be provided by September 24, 2008.

Question 15.01.01 - 15.01.04-12:

Please clarify why SG-3 has higher heat transfer (Fig. 15.1-36) yet a higher outlet temperature (Fig 15.1-37) compared to the other loops.

Regulatory basis: SRP 15.1.1-15.1.4, p. 2. "*The results of the transient analysis are reviewed....*"

Response to Question 15.01.01 - 15.01.04-12:

A response to this question will be provided by September 24, 2008.

Question 15.01.01 - 15.01.04-13:

For event addressed in Section 15.1.4, please provide information to support the conclusion that the peak return to power for each HZP case does not cause fuel damage.

Regulatory basis: SRP 15.1.1-15.1.4, p. 2. *"The results of the transient analysis are reviewed to ensure system parameters are within the ranges expected...DNBR..."*

Response to Question 15.01.01 - 15.01.04-13:

A response to this question will be provided by October 30, 2008.

Question 15.01.05-1:

Please provide the results of pipe failures inside the containment and provide justification that the 1.72 ft² break outside containment is limiting with respect to core performance.

Regulatory basis: SRP 15.1.5, p. 2. "...a range of break sizes must be considered both inside and outside containment to determine the acceptability of the system response"

Response to Question 15.01.05-1:

The main steam pressure sensors are located outside containment, and a break outside containment would expose them to harsh environmental conditions. The normal and degraded pressure uncertainties are 30 psi and 75 psi, respectively. Because of the impact of a break outside containment on the protection system response, a break outside containment generates more limiting results than a break inside containment.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.05-2:

Please provide the results of HFP and HZP cases with and without loss of offsite power so that the staff can verify the accident scenario presented is indeed limiting.

Response to Question 15.01.05-2:

A response to this question will be provided by October 30, 2008.

Question 15.01.05-3:

Please provide a discussion of the pre-scrum portion of the MSLB.

Response to Question 15.01.05-3:

A response to this question will be provided by October 30, 2008.

Question 15.01.05-4:

The EPR is designed to have a constant T_{avg} for power levels above 60%. Explain how it was determined that intermediate power levels do not present a more challenging initial condition.

Response to Question 15.01.05-4:

A response to this question will be provided by October 30, 2008.

Question 15.01.05-5:

Please explain why the Tech. Spec. minimum flow used in the analysis is conservative.

Regulatory basis: SRP 15.1.5, p. 2. *“Evaluation with various assumed initial conditions is required to verify that the condition leading to the severest consequences has been identified”*

Response to Question 15.01.05-5:

See the response to Question 15.01.01 - 15.01.04-4.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.05-6:

Please explain why modeling the SG secondary boiler region as a single volume is conservative.

Regulatory basis: Final SER of TR ANP-10263(P), p. 12. *“The staff finds the methods that utilizes S-RELAP5 are in general applicable to the U.S. EPR; however, it has deferred its final position on the complete methodology that uses S-RELAP5 code until the justification for steam generator modeling is provided.”*

Response to Question 15.01.05-6:

This question is similar to RAI 10 regarding ANP-10263P-A, “Codes and Methods Applicability Report for the U.S. EPR.” The RAI and AREVA’s April 24, 2007 response in ANP-10263Q1NP Revision 1, are as follows:

NRC Comment

Page 5-33 of ANP-10263NP:

“Steam generators with axial economizers: The U.S. EPR steam generator design features an axial economizer which channels the feedwater through the 180° sector of the downcomer and lower boiler regions that corresponds to the cooler, downflow legs of the U-tubes. However, modeling this feature is not warranted for a Main Steam Line Break analysis, in which all of a steam generator’s downcomer and boiler regions are lumped together into a single volume and the feedwater is injected into the bottom of that volume.”

RAI 10: *This needs further clarification. Is the statement based simply on using the previous modeling approach or has it been, or will it be, demonstrated through sensitivity studies? What is done with the additional SG structures which form the economizer? Why is it not important to consider the economizer for this event?*

Response 10:

For the U.S. EPR, main feedwater is terminated early in the Main Steam Line Break (MSLB) transient and emergency feedwater is initiated. While main feedwater is injected directly into the top of the economizer, emergency feedwater is sprayed relatively high in the generator above the economizer structure. Both hot and cold sides of the downcomer receive emergency feedwater, thus effectively negating the effect of the economizer. This makes it unnecessary to model the economizer explicitly since it performs its function for only a very short period during the transient. Without the economizer, therefore, the generator is equivalent to those in typical Westinghouse or CE designs. The model used to analyze MSLB events in those plants will perform as well for the U.S. EPR.

The MSLB model for the U.S. EPR follows the approved methodology prescribed in Reference 10-1. The basis of the methodology is that fuel integrity (and primary loop integrity) is dependent on the heat removal rate from the primary. Conservative

treatment of the break - allowing steam only out the break - maximizes the heat removal from the primary. This maximizes the reactivity impact of the cooldown, leads to the greatest probability of return to power, and maximizes the effect of the cooldown on the primary side level control system (pressurizer). From this perspective, the steam generator provides a boundary condition to the primary system.

As prescribed by the MSLB methodology, the steam generator (SG) secondary side fluid volume below the top of the primary steam separators is lumped in a single control volume. The fluid volume above this point is included in a second control volume (the steam dome), connected to the lower region by a junction that allows only steam to pass to the dome. This perfect separation of the liquid and vapor phases results in the removal of more energy from the steam generator than would occur if a two-phase mixture were allowed to leave the steam generator through the break (if an explicitly modeled separator were used).

Using a single control volume to represent the lower portion of the SG is conservative for the MSLB analysis. The incoming cold feedwater is mixed with all of the liquid in the SG and produces a lower bulk secondary side fluid temperature than if the feedwater is mixed with only a portion of the SG liquid in a more complex nodalization that does not have the feedwater in contact with all of the fluid.

Sensitivity studies on the nodalization are not required because adding detail tends to impede cooldown associated with the simplified model. Note that the passive heat structures such as the SG shell wall, the tube wrapper, the tube divider plate, and the economizer shell are conservatively omitted from the model as these would release heat as the SG cools down, thus mitigating the temperature reduction.

Reference 10-1 – “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” EMF-2310(P)(A) Revision 1, May 2004.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.05-7:

Page 15.1-20 states the mixing of fluid between core sectors and conservative treatment of MFW flow are both described in reference 1 (ANP-10263P-A). Please indicate where they are described in that reference.

Response to Question 15.01.05-7:

ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR," refers to EMF 2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," for use in non-LOCA analyses. The mixing of fluid between core sectors is described in EMF-2310(P)(A), Revision 1, Section 5.4.3.3. The treatment of main feedwater (MFW) flow is described in Section 5.4.3.7 of EMF-2310(P)(A), Revision 1.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.05-8:

Please clarify section 15.1.5.1, second paragraph, where it is stated “*Because the break is assumed to be located upstream of an MSIV, their closure isolates the **affected** SG from the break (emphasis added)*”

Response to Question 15.01.05-8:

A response to this question will be provided by October 30, 2008.

Question 15.01.05-9:

Please explain the apparent inconsistency of Tables 15.1-15 and 15.1-16. The former says the power peaks at 23.14% of RTP while the latter says the power peaks at 649.28 MWt, which is 14.15% of RTP.

Response to Question 15.01.05-9:

A response to this question will be provided by October 30, 2008.

Question 15.01.05-10:

Please explain how the fuel failure at clad strain limit value (Table 15.1-16) is calculated.

Response to Question 15.01.05-10:

An evaluation of the peaking factors of all the assemblies within the core determines the number of failed assemblies. A limiting peaking factor is calculated using the linear heat generation rate (LHGR) cladding strain limit and the event-specific average core peak LHGR. The assemblies that have a peaking factor above this calculated limited are considered failed.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.05-11:

Please confirm that the initial shutdown margin presented in Table 15.1-13 is the minimum shutdown margin (SDM) allowed by the Technical Specifications.

Response to Question 15.01.05-11:

The shutdown margin presented in U.S. EPR FSAR, Tier 2, Table 15.1-13—MSLB - Key Input Parameters for Limiting Case, 3,000 percent millirho (pcm), corresponds to the minimum shutdown margin required by the core operating limits report referenced in the Technical Specifications.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.01.05-12:

Please explain the reactor power increase shown in Figure 15.1-55.

Response to Question 15.01.05-12:

A response to this question will be provided by October 30, 2008.

Question 15.02.01-15.02.05-1:

Section 15.2.2.4 states that the radiological consequences of the turbine trip event are bounded by the inadvertent opening of an MSSV event described in Section 15.1.4. The approach of comparing one event to another in different event type (Heat up transients vs cooldown transients) requires more justifications since the assumptions, initial conditions and other plant conditions for different type of events will not provide same base line for comparisons. For each type of events, please identify the most limiting case with respect to all acceptance criteria within the group of events in the same event type.

Note: The approach of comparing events with different types has been used in many cases in the EPR DC application.

Regulatory basis: SRP acceptance criteria instructs the reviewer to verify "*whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.*"

Response to Question 15.02.01-15.02.05-1:

A response to this question will be provided by October 30, 2008.

Question 15.02.01-15.02.05-2:

The turbine trip transient listed in Table 15.2.3 assumed with LOOP at 7.37 sec. Turbine trip with offsite power available may result in higher primary system pressure. Please provide an analysis with and without loss offsite power available for each transient and accident. Explain the physical basis that leads to one sequence being more limiting than the other.

SRP 15.2.1-15.2.5 states "To the extent deemed necessary, the reviewer evaluates the effect of single active system or component failures that may affect the course of the transient. For new applications, LOOP should not be considered a single failure; each of the reduction-of-heat-removal transients should be analyzed with and without a LOOP in combination with a single active failure.

Response to Question 15.02.01-15.02.05-2:

A response to this question will be provided by October 30, 2008.

Question 15.02.01-15.02.05-3:

In Section 15.2.2, the analysis assumes single failure of a secondary side MSRIV. All 3 pressurizer safety valves are assumed to function. With respect to maximum primary pressure, explain why the secondary side safety valve failure to open is more restrictive than a primary side safety valve failure to open.

SRP acceptance criteria instructs the reviewer to verify “whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.

RG 1.206, C.I.15.6.2 *Sequence of Events and Systems Operation* states “The applicant should discuss the following considerations for each initiating event: ...List all single failures or operator errors considered in the transient and accident analysis, and identify the limiting single failure for each event.

Response to Question 15.02.01-15.02.05-3:

The pressurizer safety relief valves (PSRVs) are single-failure proof, whereas the main steam relief isolation valves (MSRIVs) are not. U.S. EPR FSAR, Tier 2, Section 5.2.2.9 notes:

5.2.2.9 System Reliability

During hot RCS conditions, the PSRVs are considered a passive device. The spring operated pilot valves are designed in accordance with the requirements of ASME Section III, NB-7511.1. With successful operation of the pilot valve, a large differential pressure reliably opens the main relief disk, which relieves RCS pressure.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.02.01-15.02.05-4:

Please provide the minimum DNBR as a function of time.

SRP 15.2.1-15.2.5-5 Revision 2 - March 2007 states the reviewer confirms that:

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

B. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs.

Response to Question 15.02.01-15.02.05-4:

A response to this question will be provided by October 30, 2008.

Question 15.02.01-15.02.05-5:

In the event addressed in Section 15.2.4,,explain the prediction of MSIV flow rate oscillation shown in Figure 15.2-MSIVC Secondary Overpressurization—MSIV Flow Rates. It appears that the oscillations are due to rapid closure of the turbine valve. Pressure waves bouncing back and forth in the steam lines?

Response to Question 15.02.01-15.02.05-5:

The oscillations in the three unaffected steam lines are due to the fast closure of the turbine valve. The main steam isolation valve (MSIV) in steam line 1 is assumed to fail closed in the MSIV closure (MSIVC) analysis. The fast closure of the turbine valve results in the pressure waves reverberating in the three unaffected steam lines.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.02.01-15.02.05-6:

Section 15.2.4.3.2 DNBR analysis results do not provide the results of the DNBR calculation. Please provide a figure showing minimum DNBR as a function of time.

Response to Question 15.02.01-15.02.05-6:

A response to this question will be provided by October 30, 2008.

Question 15.02.01-15.02.05-7:

Please provide a description of how DNBR calculations were done for the event addressed in Section 15.2.4.

Response to Question 15.02.01-15.02.05-7:

A response to this question will be provided by October 30, 2008.

Question 15.02.01-15.02.05-8:

Section 15.2.4.3.2 implies that a 3-D kinetics calculation was done. This would mean S-RELAP5 would have to have sectorized vessel in order to get asymmetric inlet conditions to the core inlet. Please explain how the 3-D kinetics calculation was performed.

Response to Question 15.02.01-15.02.05-8:

A response to this question will be provided by October 30, 2008.

Question 15.02.06-1:

Insufficient information is provided to assess the conclusions in the FSAR. The FSAR states that this event is bounded by the turbine trip event. Provide a sequence of events explaining the functioning of the reactor coolant pumps, feedwater pumps, and other auxiliaries which are lost upon loss of non-emergency power and explain why this event is bounded.

SRP 15.2.6 III. REVIEW PROCEDURES states "2. If the SAR states that the loss of ac power transient is not as limiting as some other similar transient, the reviewer evaluates the applicant's justification.

Review instructions include "The sequence of events from initiation until condition stabilization is reviewed to ascertain:

- A. The extent to which normally operating plant instrumentation and controls are assumed to function.
- B. The extent to which plant and reactor protection systems are required to function.
- C. The credit taken for the functioning of normally operating plant systems.
- D. The operation of engineered safety systems required.
- E. The extent to which operator actions are required.
- F. The operation of standby diesel generators required."

Response to Question 15.02.06-1:

A response to this question will be provided by October 30, 2008.

Question 15.02.07-1:

Please provide a figure showing minimum DNBR as a function of time for this event.

Response to Question 15.02.07-1:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-1:

Provide description of the modeling methodology used to analyze the reactor behavior during feedwater pipe break transients.

Response to Question 15.02.08-1:

ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR" (specifically section 5.2.2.2.3) and EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," describe the applicability and general methodology used for the feedwater line break analysis.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.02.08-2:

Provide data of calculated transient DNBR and amount of fuel failures to support the evaluation of the acceptance criteria for this event as stated in Section 15.2.8.2 of the FSAR.

Response to Question 15.02.08-2:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-3:

Figure 15.2-67—FWLB Representative Small Break – “Reactivities shows liquid fraction vs time”. Figure 15.2-68—“FWLB Representative Small Break – Liquid Volume Fraction in Pressurizer Dome” shows power vs time. Figure 15.2-69—“FWLB Maximum RCS Pressure Case – Reactor and Total Steam Generator Power shows pressure vs time”. Please explain the discrepancies between the titles and the parameters presented in the transient curves.

Response to Question 15.02.08-3:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-4:

For transients analyzed in Sections 15.2.1 through 15.2-8, provide the decay heat curve used to calculate the total core thermal power. Specify the model used to calculate the decay power curve along with any assumptions made. Use a logarithmic scale over an appropriate range in displaying this quantity (SRP 15.2.1-15.2.5 Section III.6.A, 15.2.6 Section III.5.A, 15.2.7-15.2.8 Section III)

Response to Question 15.02.08-4:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-5:

Figure 15.2-50 “FWLB Representative Small Break - Pressurizer Pressure” shows a predicted pressure range, which is above 1,600 psia till the end point of analysis at 7,000 s. Figure 15.2-65 “FWLB Representative Small Break – RCS Maximum Pressure” reveals a predicted pressure range that is below 1,500 psia from the beginning of the analysis at 0 s. Please explain the difference (15.2.8 Section III).

Response to Question 15.02.08-5:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-6:

Figure 15.2-66 “FWLB Representative Small Break – Steam Generator Maximum Pressure” shows reactivity quantities plotted in units of \$. Please clarify the unit. (15.2.8 Section III).

Response to Question 15.02.08-6:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-7:

Figure 15.2-67 “FWLB Representative Small Break – Reactivities” shows a liquid fraction quantity in dimensionless units. Please clarify the unit. (15.2.8 Section III).

Response to Question 15.02.08-7:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-8:

Figure 15.2-68 “Representative Small Break – Liquid Volume Fraction in Pressurizer Dome” shows quantities in units of MW. Please clarify the unit. (15.2.8 Section III).

Response to Question 15.02.08-8:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-9:

For the break flow in Figure 15.2-58 “FWLB Representative Small Break – Break Flow”, explain how the break flow area was determined. Please explain the criterion used for determining the worst break size. Provide the flow quality as well as the fluid-phase temperatures at the break opening. In addition, provide the liquid- and vapor-phase mass flow rate at the break and specify the break backpressure used in the calculations. Use a logarithmic scale over an appropriate range in displaying quantities that vary significantly in magnitude (15.2.8 Section III).

Response to Question 15.02.08-9:

A spectrum of break sizes is investigated that ranges from the minimum break size that could be overfed by the main feedwater (MFW), 0.0184 ft², to the area of the MFW distribution half-ring inlet, 0.922 ft². The representative small break results presented produce the maximum pressurizer level.

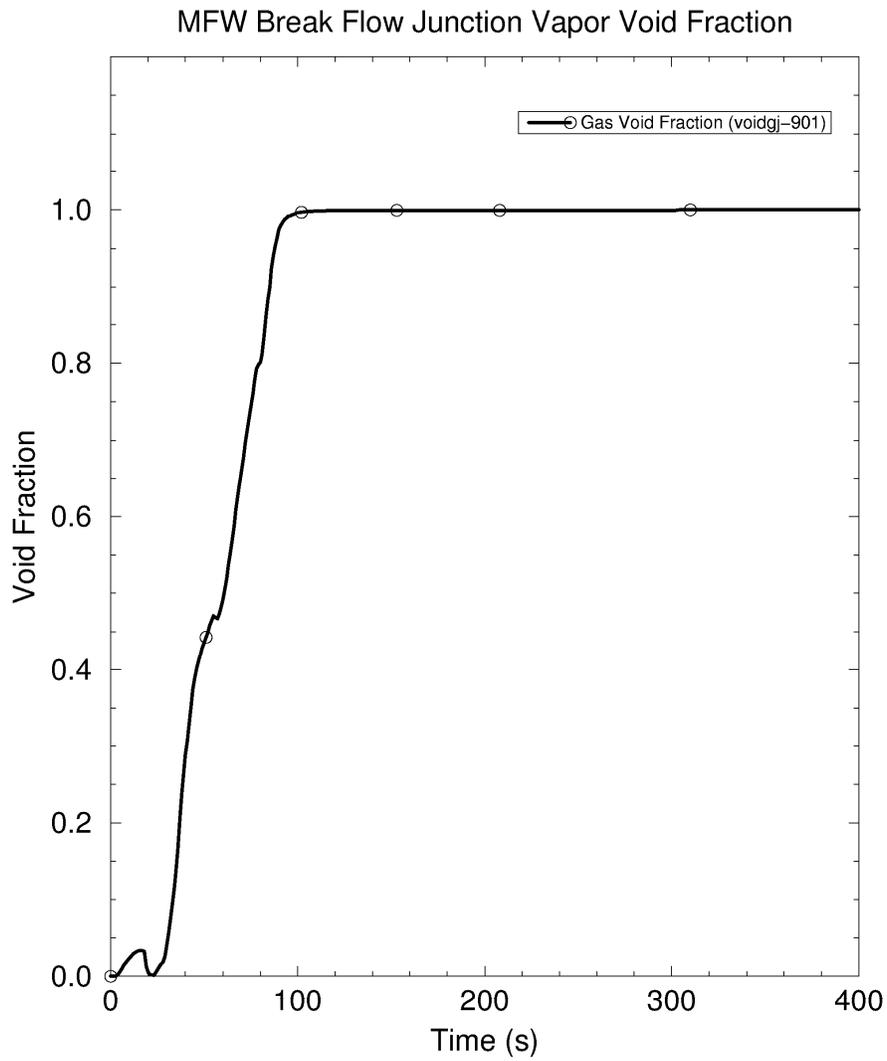
The break flow is determined by S-RELAP5 utilizing the single velocity Moody critical break flow correlation with a forward loss coefficient of 1.0 and a reverse loss coefficient of 0.5. The containment pressure (i.e., break backpressure) is fixed at 14.7 psia.

The void fraction at the break opening is provided in attached Figure 15.02.08-9-1 (time scale to 400 seconds) and Figure 15.02.08-9-2 (time scale to 7000 seconds). The vapor phase temperature and saturation temperature in the break upstream volume is provided in attached Figure 15.02.08-9-3 (time scale to 400 seconds) and Figure 15.02.08-9-4 (time scale to 7000 seconds). The liquid and vapor mass flows are provided in Figure 15.02.08-9-5 (time scale to 400 seconds) and Figure 15.02.08-9-6 (time scale to 7000 seconds).

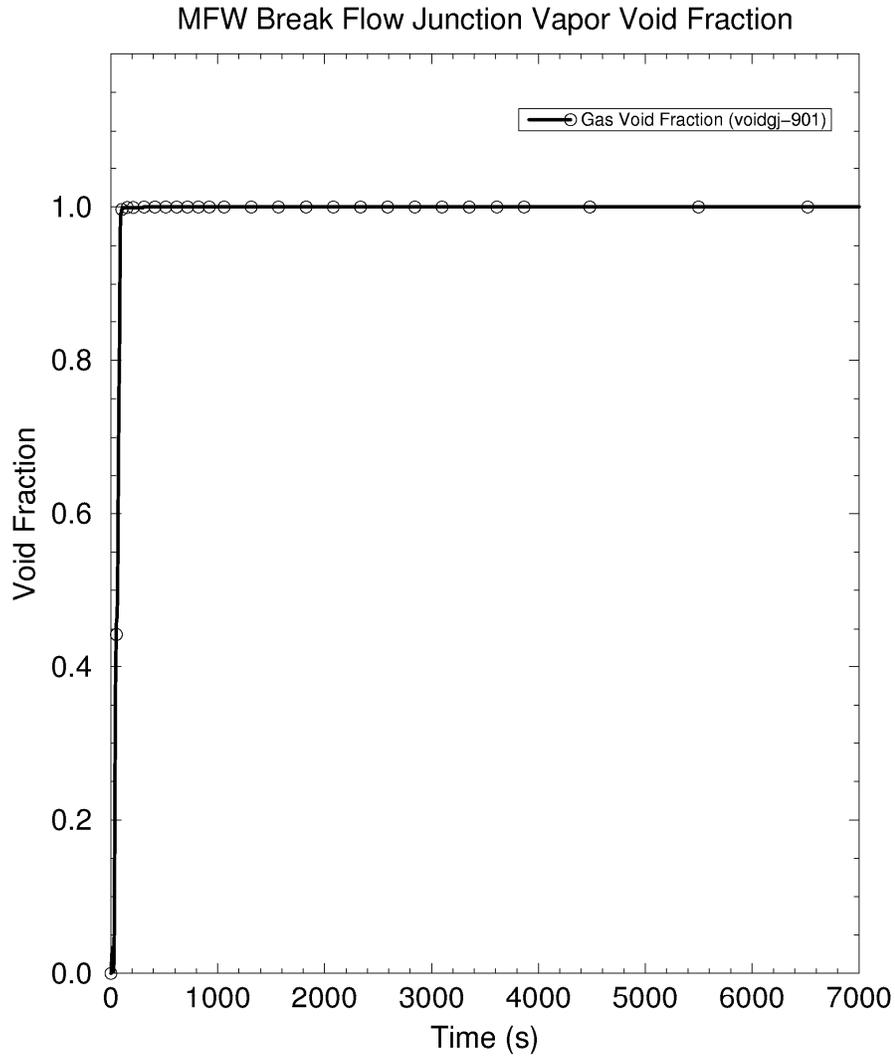
FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

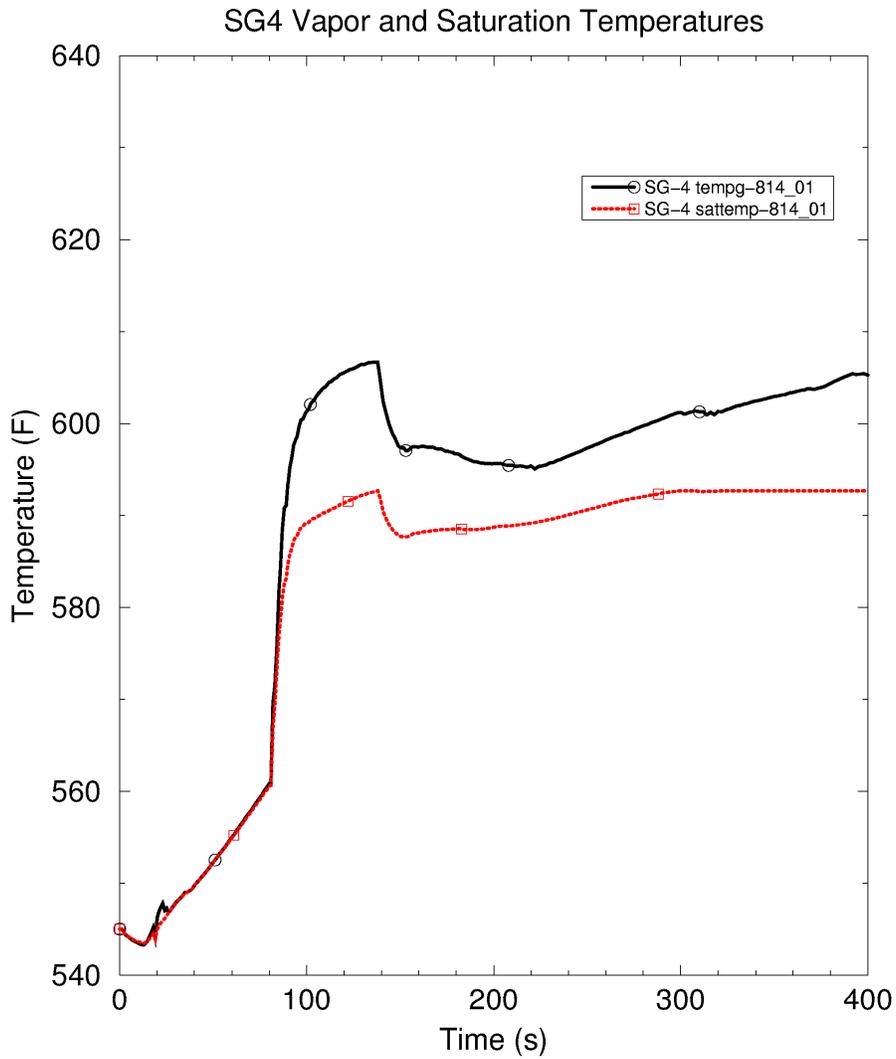
15.02.08-9-1—FWLB Representative Small Break – Vapor Void Fraction at the Break



15.02.08-9-2—FWLB Representative Small Break – Vapor Void Fraction at the Break



15.02.08-9-3—FWLB Representative Small Break – Break Upstream Volume Vapor and Saturation Temperature



15.02.08-9-4—FWLB Representative Small Break – Break Upstream Volume Vapor and Saturation Temperature

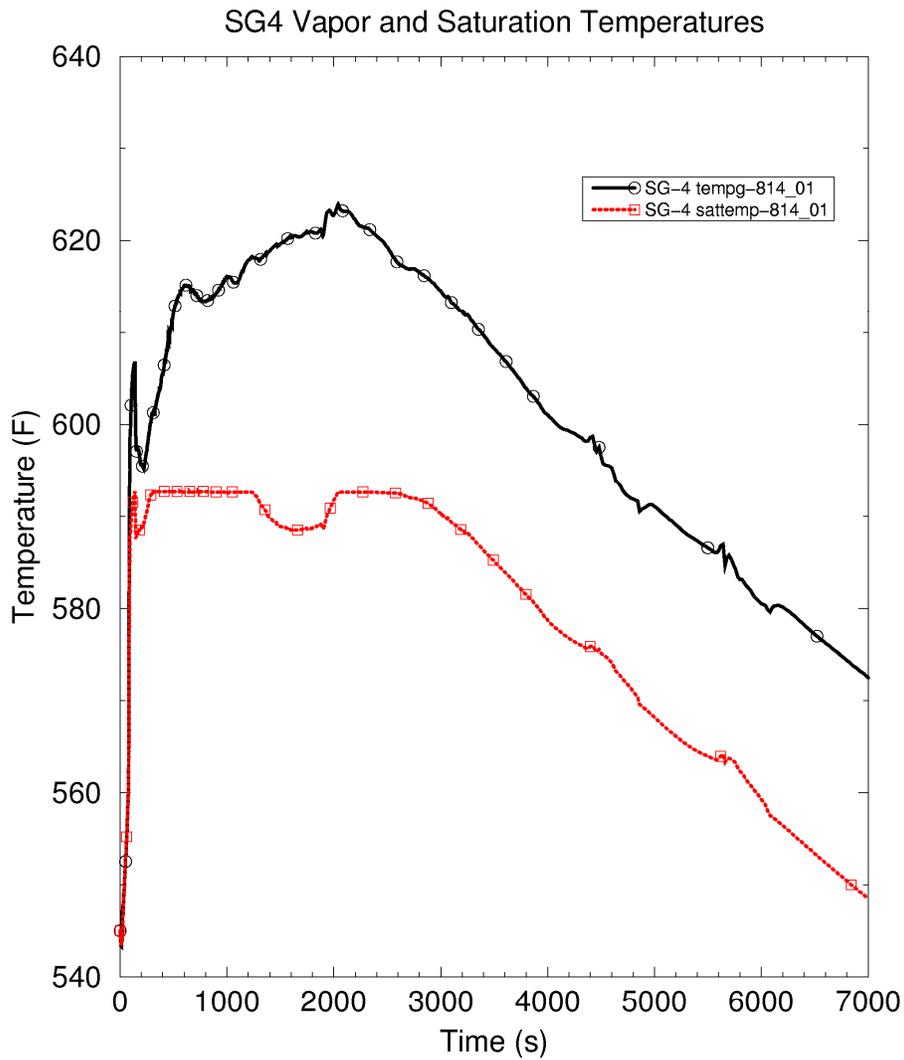
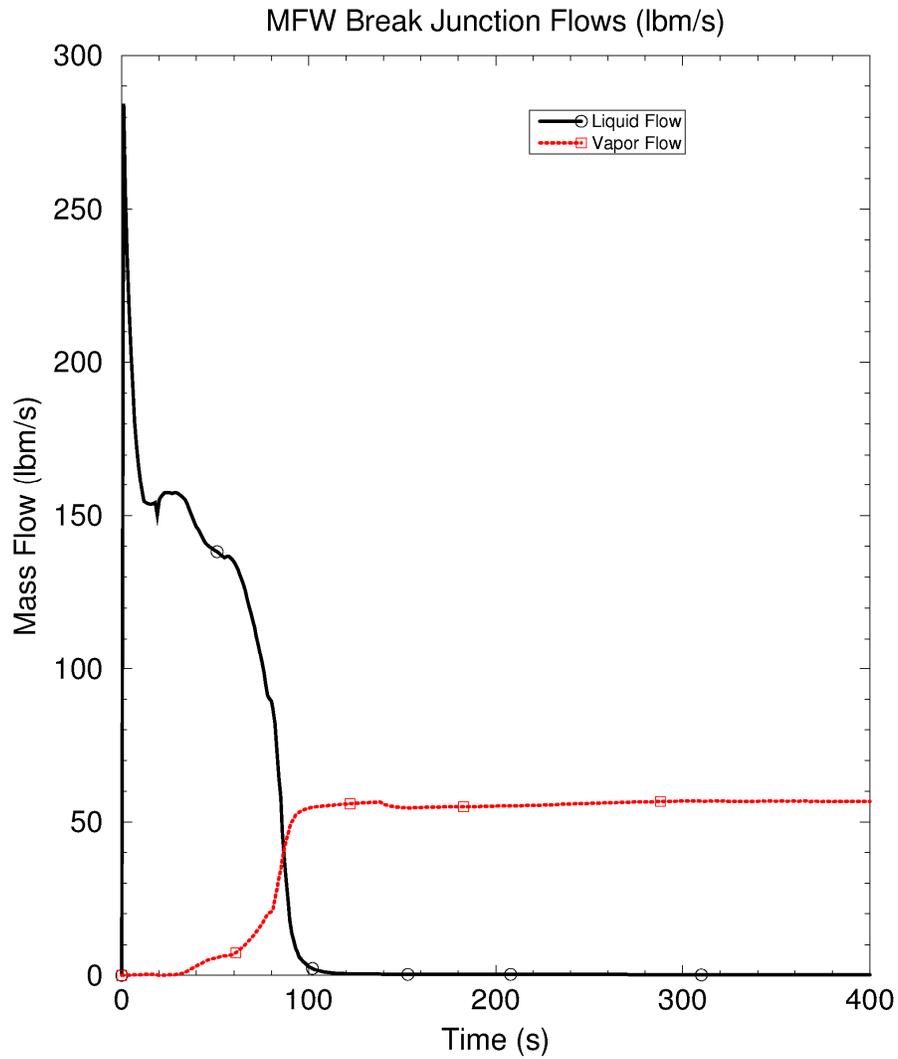
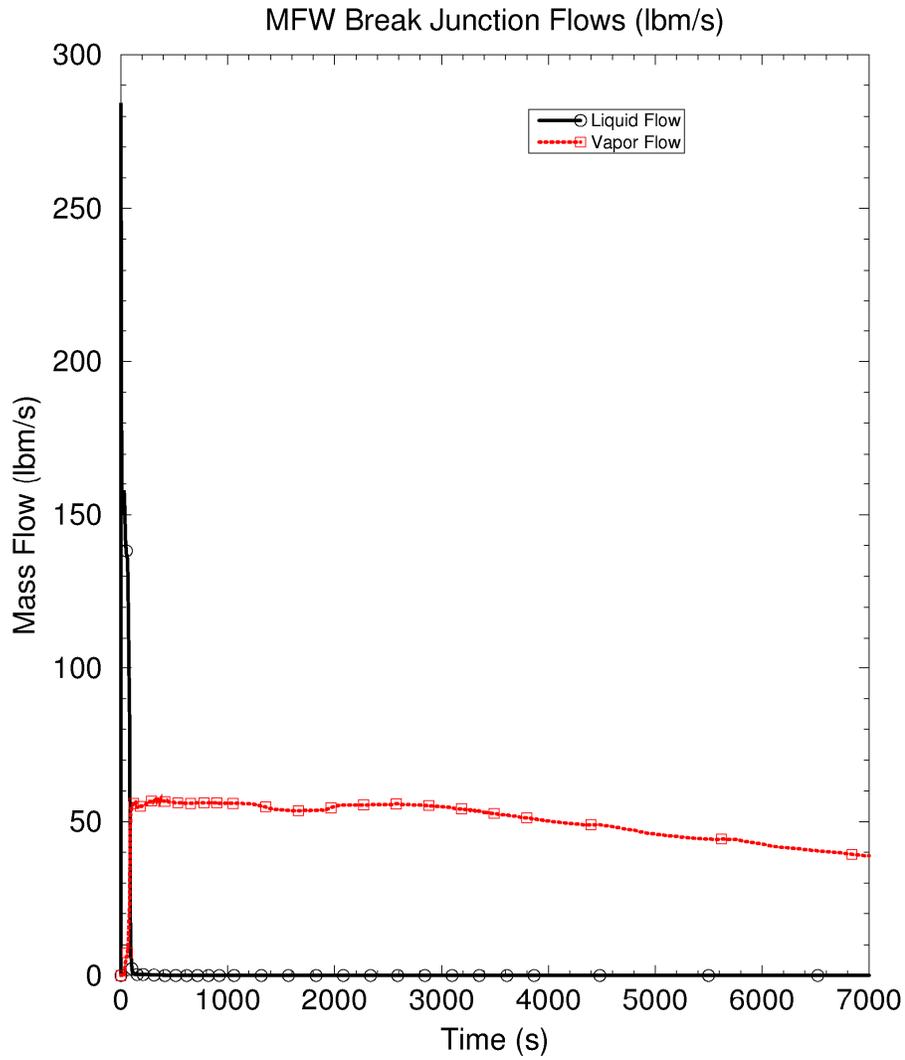


Figure 15.02.08-9-5—FWLB Representative Small Break – Break Phase Flows



15.02.08-9-6—FWLB Representative Small Break – Break Phase Flows



Question 15.02.08-10:

Figure 15.2-69 “FWLB Maximum RCS Pressure Case – Reactor and Total Steam Generator Power” shows a quantity in psia units. Please clarify the unit. (15.2.8 Section III).

Response to Question 15.02.08-10:

A response to this question will be provided by October 30, 2008.

Question 15.02.08-11:

For the break flow in Figure 15.2-77 "FWLB Maximum RCS Pressure Case – Break Flow" explain how the break flow area was determined. Please explain the criterion used for determining the worst break size. Demonstrate that it produces the highest peak RCS pressure. Provide the flow quality as well as the fluid-phase temperatures at the break opening. In addition, provide the liquid- and vapor-phase mass flow rate at the break and specify the break backpressure used in the calculations. Use a logarithmic scale over an appropriate range in displaying quantities that vary significantly in magnitude (15.2.8 Section III).

Response to Question 15.02.08-11:

A spectrum of break sizes is investigated that ranges from the minimum break size that could be overfed by the main feedwater (MFW), 0.0184 ft², to the area of the MFW distribution half-ring inlet, 0.922 ft². The results of the break size cases are reviewed to determine which break size produced the maximum reactor coolant system (RCS) pressure. The limiting RCS pressure case is the 0.415 ft² break (45 percent of the largest break size investigated).

The break flow is determined by S-RELAP5 utilizing the single velocity Moody critical break flow correlation with a forward loss coefficient of 1.0 and a reverse loss coefficient of 0.5. The containment pressure (i.e., break backpressure) is fixed at 14.7 psia.

The void fraction at the break opening is provided in the attached Figure 15.02.08-11-1 (time scale to 400 seconds) and Figure 15.02.08-11-2 (time scale to 7000 seconds). The vapor phase temperature and saturation temperature in the break upstream volume are provided in the attached Figure 15.02.08-11-3 (time scale to 400 seconds) and Figure 15.02.08-11-4 (time scale to 7000 seconds). The liquid and vapor mass flows are provided in Figure 15.02.08-11-5 (time scale to 400 seconds) and Figure 15.02.08-11-6 (time scale to 7000 seconds).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Figure 15.02.08-11-1—FWLB Maximum RCS Pressure Case – Vapor Void Fraction at the Break

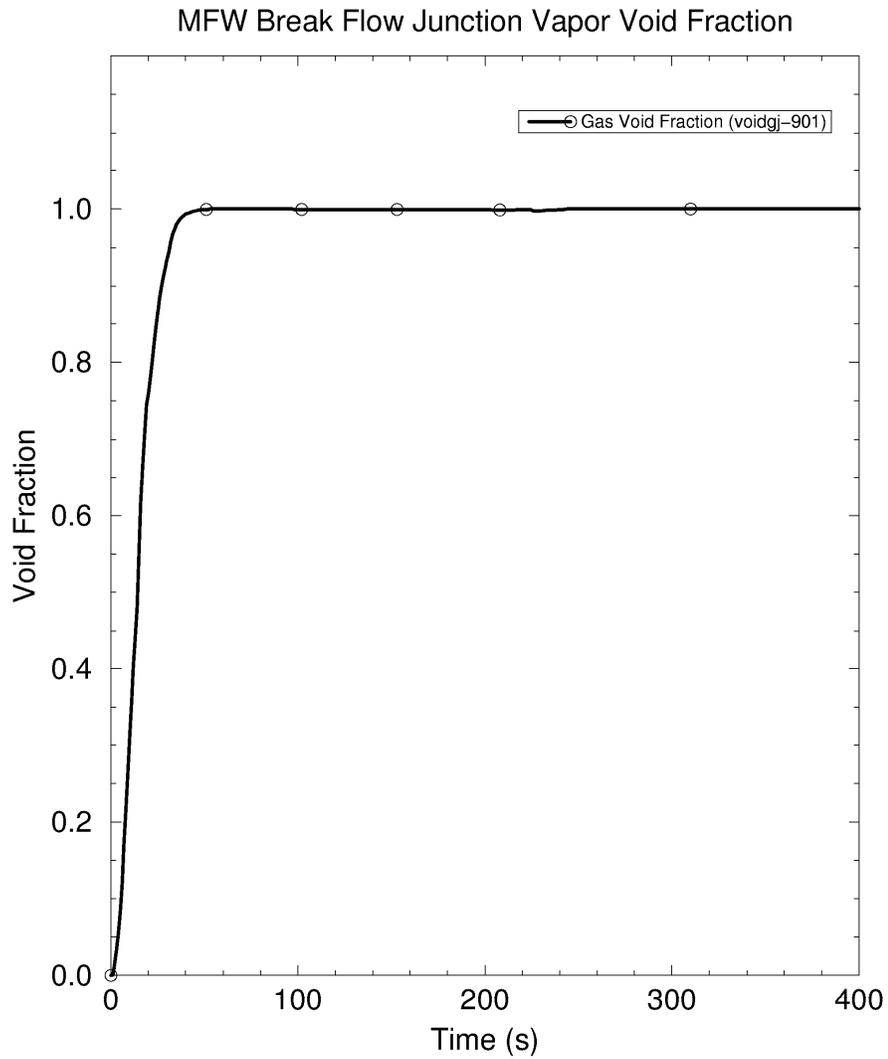
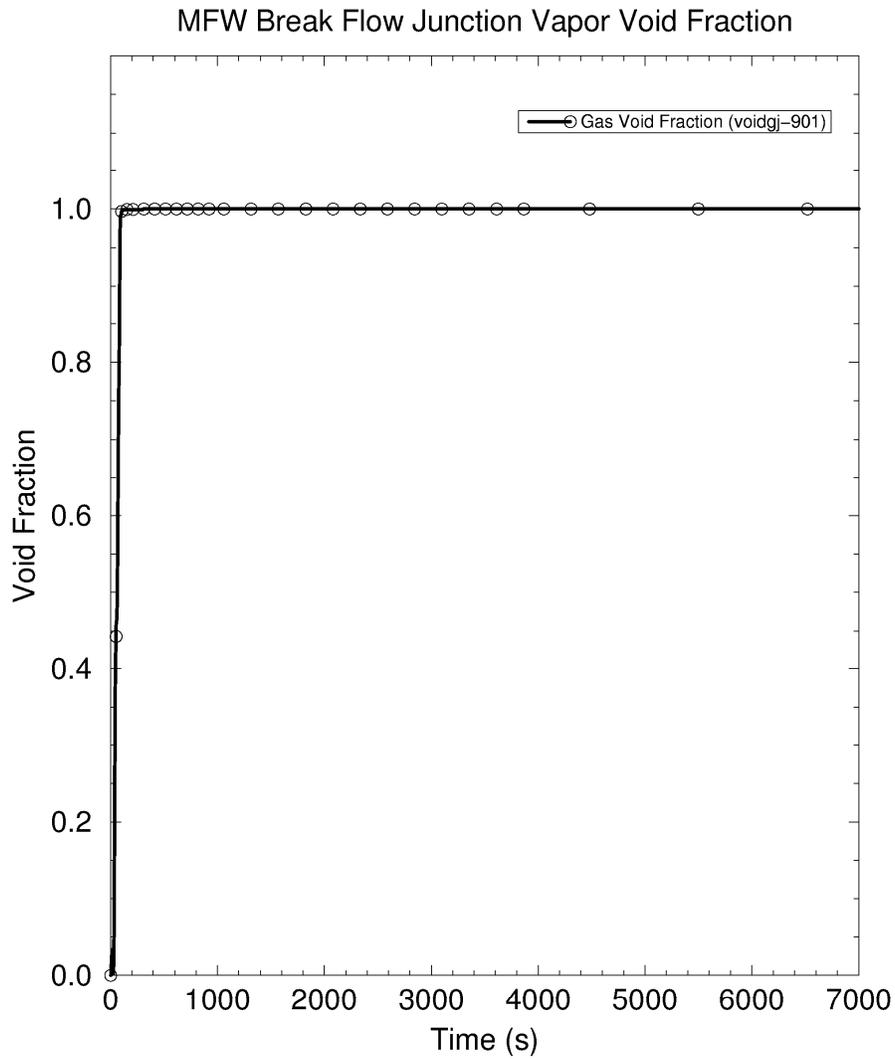
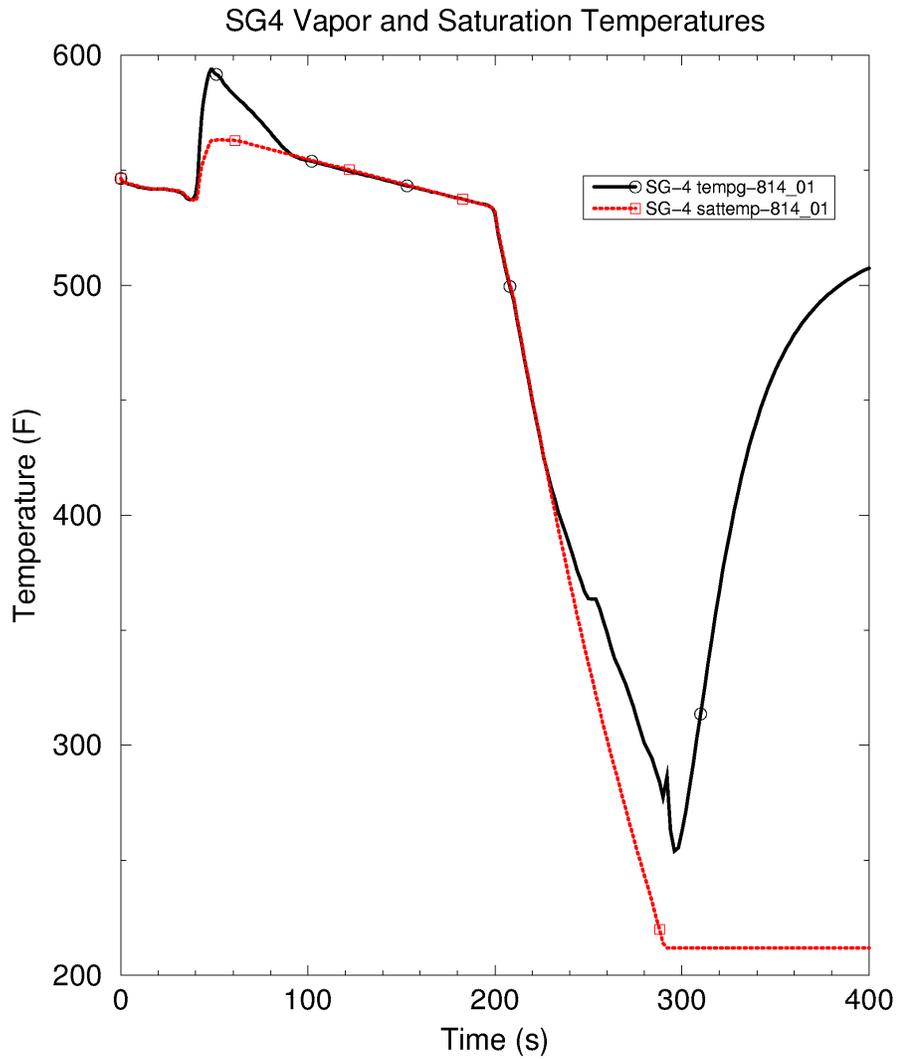


Figure 15.02.08-11-2—FWLB Maximum RCS Pressure Case – Vapor Void Fraction at the Break



**Figure 15.02.08-11-3—FWLB Maximum RCS Pressure Case – Break
Upstream Volume Vapor and Saturation Temperature**



**Figure 15.02.08-11-4—FWLB Maximum RCS Pressure Case – Break
Upstream Volume Vapor and Saturation Temperature**

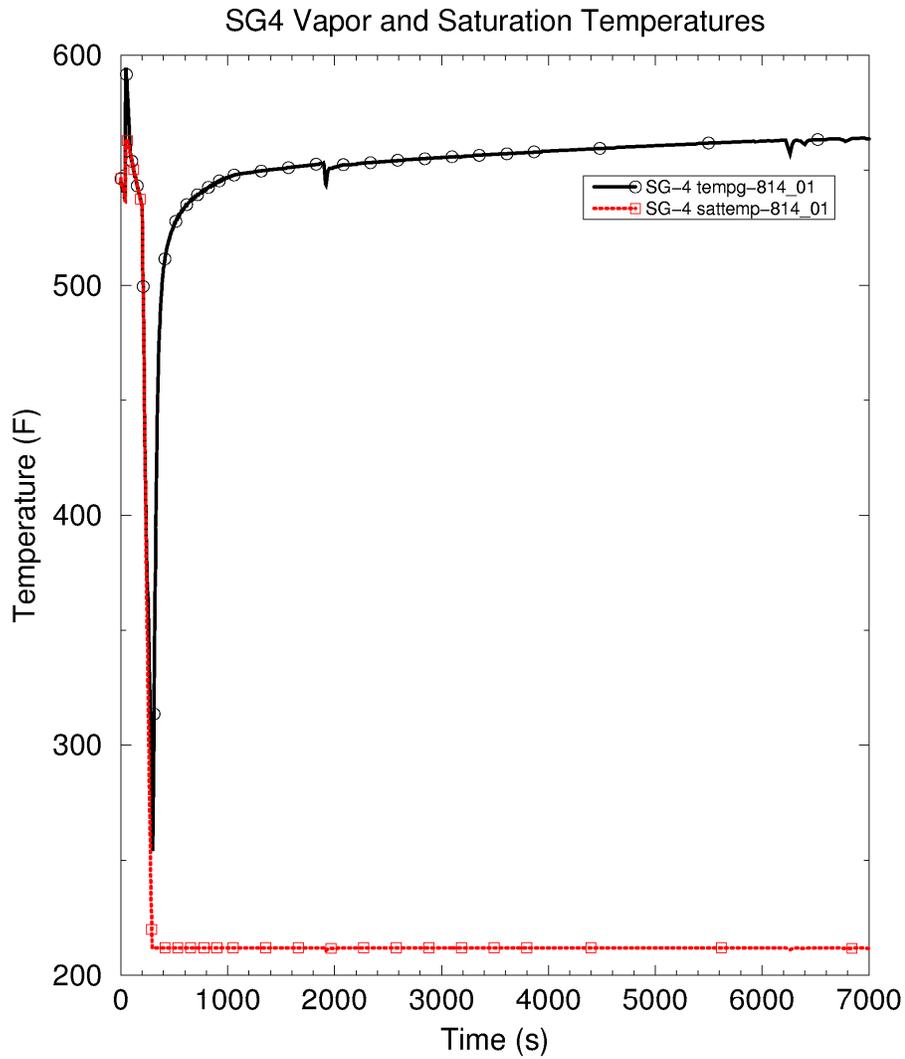


Figure 15.02.08-11-5—FWLB Maximum RCS Pressure Case – Break Phase Flows

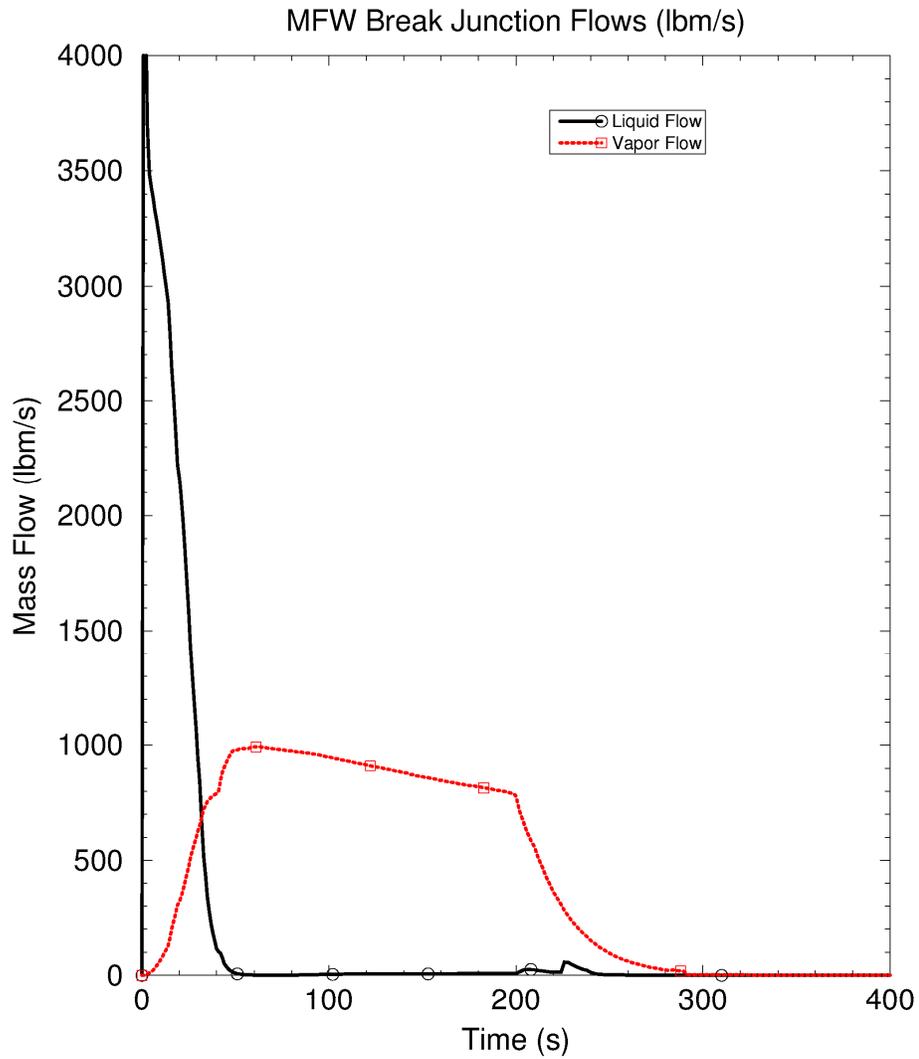
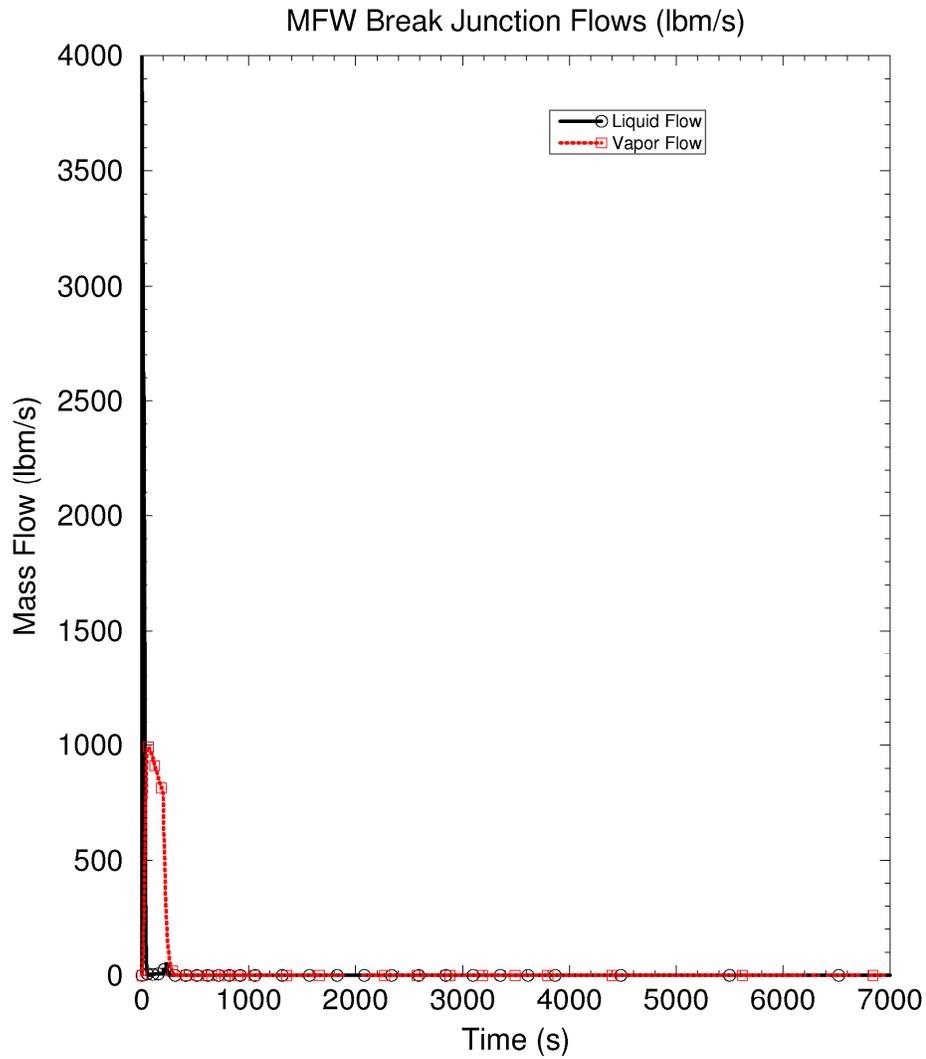


Figure 15.02.08-11-6—FWLB Maximum RCS Pressure Case – Break Phase Flows



Question 15.02.08-12:

Provide the valve flow areas used to compute the mass flow rates shown in Figure 15.2-78 “FWLB Maximum RCS Pressure Case – Main Steam Relief Loops 1 and 2” and in Figure 15.2-79 “FWLB Maximum RCS Pressure Case – Main Steam Relief Loops 3 and 4” (15.2.8 Section III).

Response to Question 15.02.08-12:

The normalized main steam relief control valve (MSRCV) flow areas are provided in Figure 15.02.08-12-1 (time scale to 400 seconds) and Figure 15.02.08-12-2 (time scale to 7000 seconds). The modeled valve full-open area for the normalization is 0.2598 ft². The main steam relief isolation valve (MSRIV) must also be opened for flow to occur in the main steam relief train (MSRT).

The MSRCV power-dependent position has a minimum value of 40 percent open from 20 percent thermal power down to zero percent power. The power-dependent position is in effect when the MSRIV is closed. When the MSRIV is open, the MSRCV position is automatically adjusted to control steam generator pressure.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

**Figure 15.02.08-12-1—FWLB Maximum RCS Pressure Case – MSRCV
Normalized Valve Flow Area**

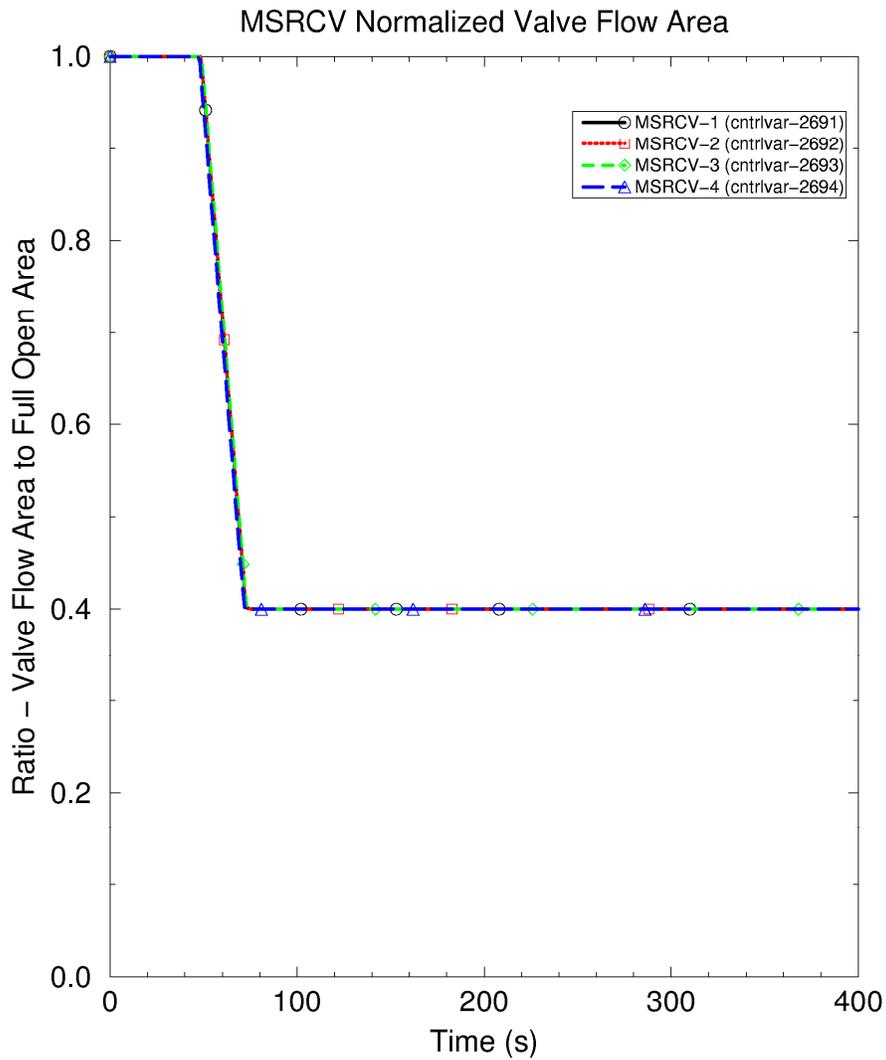
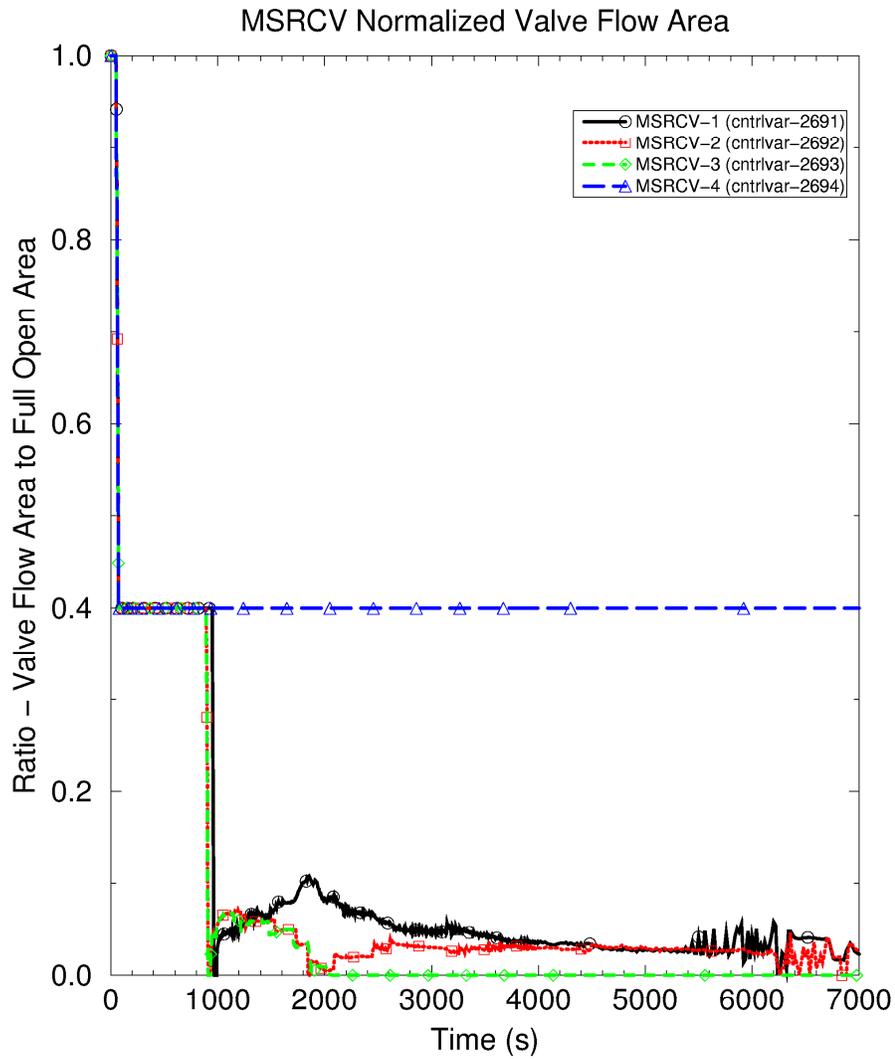


Figure 15.02.08-12-2—FWLB Maximum RCS Pressure Case – MSRCV Normalized Valve Flow Area



Question 15.02.08-13:

Provide the valve flow areas used to compute the mass flow rates shown in Figure 15.2-96 "FWLB Maximum Secondary Pressure Case – Main Steam Relief Loops 1 and 2" and in Figure 15.2-97 "FWLB Maximum Secondary Pressure Case – Main Steam Relief Loops 3 and 4" (15.2.8 Section III).

Response to Question 15.02.08-13:

The normalized main steam relief control valve (MSRCV) flow areas are provided in Figure 15.02.08-13-1 (time scale to 250 seconds) and Figure 15.02.08-13-2 (time scale to 2000 seconds). The modeled valve full-open area for the normalization is 0.2598 ft². The main steam isolation valve (MSRIV) must also be opened for flow to occur in the main steam relief train (MSRT). MSRIV-3 (for steam generator 3) is assumed to fail to open for this case, so the MSRCV-3 position has no effect on the results.

The MSRCV power-dependent position has a minimum value of 40 percent open from 20 percent thermal power down to zero percent power. The power-dependent position is in effect when the MSRIV is closed. When the MSRIV is open, the MSRCV position is automatically adjusted to control steam generator pressure.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Figure 15.02.08-13-1—FWLB Maximum Secondary Pressure Case – MSRCV Normalized Valve Flow Area

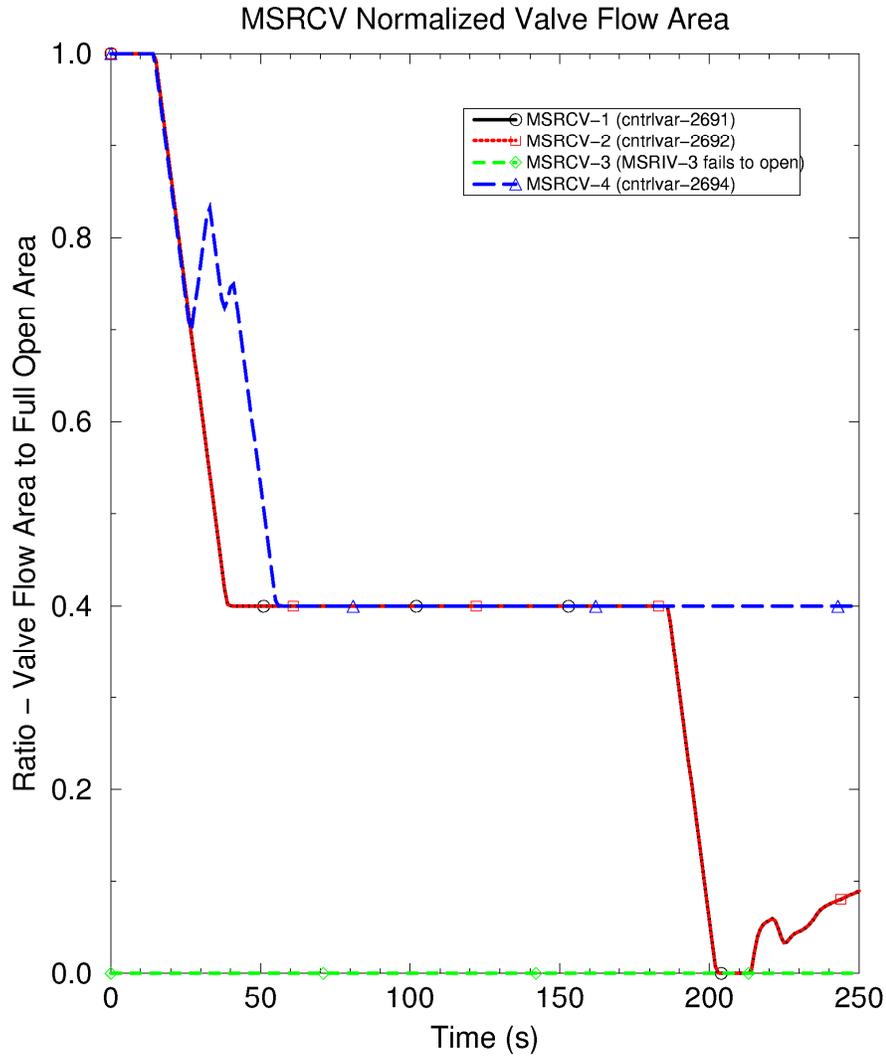
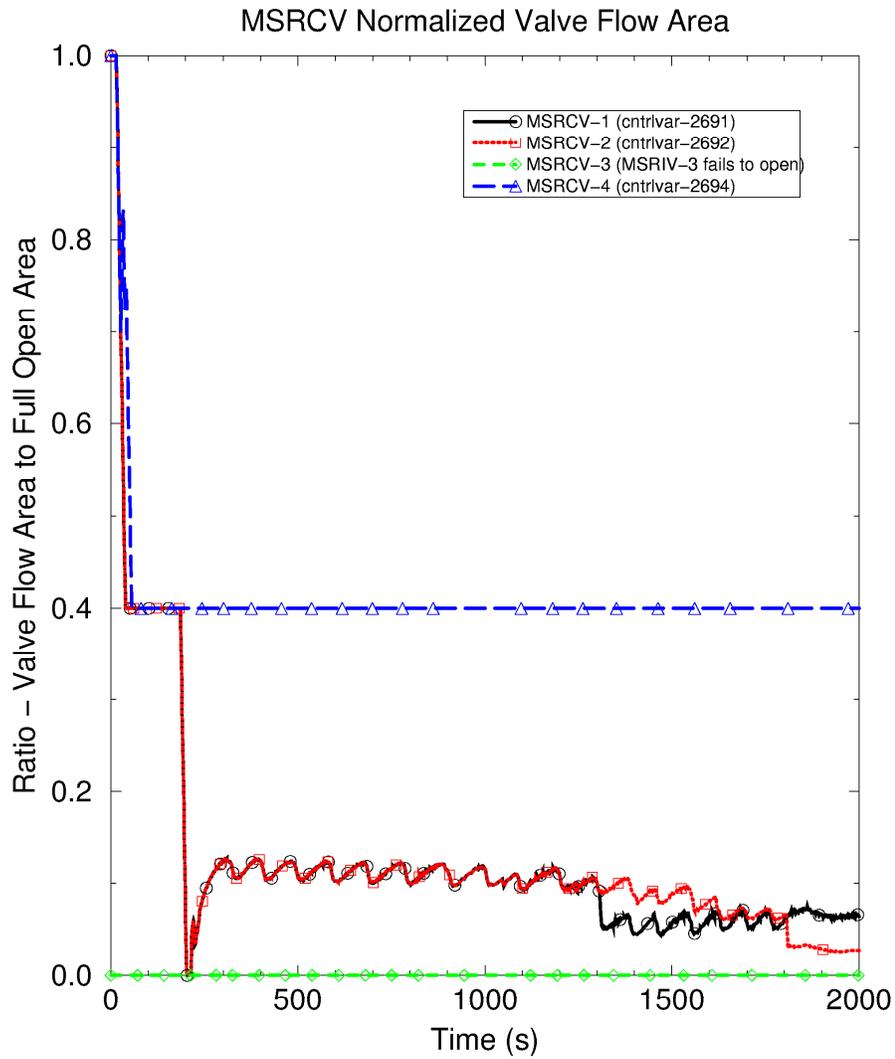


Figure 15.02.08-13-2— FWLB Maximum Secondary Pressure Case – MSRCV Normalized Valve Flow Area



Question 15.02.08-14:

Provide, as function of time, all individual reactivity components used to compute the total reactivity parameters shown in Figures 15.2.5 and 15.2-22 (15.2.8 Section III).

Response to Question 15.02.08-14:

A response to this question will be provided by October 30, 2008.

Question 15.03.03-15.03.04-1:

Provide plots showing the total reactivity as well as each individual reactivity component used in computing the reactor power for the reactor transients analyzed in Sections 15.3.1 through 15.3.4. Provide those quantities as function of time (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III).

Response to Question 15.03.03-15.03.04-1:

A response to this question will be provided by October 30, 2008.

Question 15.03.03-15.03.04-2:

Quantify the RCP inertia applied for the reactor transients in Sections 15.3.1 through 15.3.4 and explain any assumptions used in determining the inertia for all individual RCPs (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III)

Response to Question 15.03.03-15.03.04-2:

The minimum reactor coolant system flow rate as a function of time for a four-pump coastdown is controlled as a Tier 1 item (U.S. EPR FSAR, Tier 1, Table 2.2.1-4—Minimum Flow (% of Initial Flow) During Four Pump Coastdown) and will be confirmed by pre-operational testing. These reactor coolant pump coastdown flow characteristics are developed by reducing the minimum rotating inertia (U.S. EPR FSAR, Tier 2, Table 5.4-1—Reactor Coolant Pump Design Data) and are utilized for all of the U.S. EPR FSAR, Tier 2, Section 15.3 events.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.03.03-15.03.04-3:

Provide the RCP coast-down flow characteristics used for determining the mass flow rate in the primary reactor coolant recirculation loops for the reactor transients in Sections 15.3.1 through 15.3.4. Discuss how those characteristics were determined and explain any assumptions made (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III).

Response to Question 15.03.03-15.03.04-2:

The reactor coolant pump coastdown flow characteristics are provided in U.S. EPR FSAR, Tier 1, Table 2.2.1-4—Minimum Flow (% of Initial Flow) During Four Pump Coastdown. See the response to Question 15.03.03 - 15.03.04-2.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.03.03-15.03.04-4:

Explain how the core inlet temperature is computed and used in the determination of the DNBR criterion for the reactor transients presented in Sections 15.3.1 through 15.3.4. Provide a plot of this parameter for each individual transient (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III).

Response to Question 15.03.03-15.03.04-4:

A response to this question will be provided by October 30, 2008.

Question 15.03.03-15.03.04-5:

Provide a plot that shows the computed DNBR criterion as well as the DNB margin available as function of time for the reactor transients presented in Section 15.3.1 through 15.3.4 (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III).

Response to Question 15.03.03-15.03.04-5:

A response to this question will be provided by October 30, 2008.

Question 15.03.03-15.03.04-6:

Provide a comparison between the maximum core local heat flux and the core average heat flux shown in Figures 15.3-3, 15.3-7 and 15.3-12 (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III).

Response to Question 15.03.03-15.03.04-6:

A response to this question will be provided by October 30, 2008.

Question 15.03.03-15.03.04-7:

Plot governing thermal hydraulic quantities that enter into the DNBR calculation such as, but not limited to, local mass flow rate and heat flux for the reactor transients in Sections 15.3.1 through 15.3.4 (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III). Explain the applicability of the DNBR correlation to the range of the relevant parameters.

Response to Question 15.03.03-15.03.04-7:

A response to this question will be provided by October 30, 2008.

Question 15.03.03-15.03.04-8:

Plot the peak fuel centerline temperature as function of time for the reactor transients in Sections 15.3.1 through 15.3.4 (SRP 15.3.1-15.3.2 Section III, SRP 15.3.3-15.3.4 Section III). Explain the associated safety limit.

Response to Question 15.03.03-15.03.04-8:

A response to this question will be provided by October 30, 2008.

Question 15.06.01-1:

Provide the DNBR as a function of time for all Chapter 15.6 analyses for which it is a key parameter.

Regulatory basis: SRP 15.0, p. 15.0-10. *"The reviewer ensures that the applicant has presented the results of the analyses, including key parameters as a function of time..."*

Response to Question 15.06.01-1:

A response to this question will be provided by October 30, 2008.

Question 15.06.01-2:

For each Chapter 15.6 analysis, provide an assessment showing why the parameters, initial conditions and single failures assumed for each event are those which lead to the most conservative results.

Regulatory basis: SRP acceptance criteria instructs the reviewer to verify “*whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines*”.

RG 1.206, C.I.15.6.2 *Sequence of Events and Systems Operation* states
“*The applicant should discuss the following considerations for each initiating event:..*
“*...List all single failures or operator errors considered in the transient and accident analysis, and identify the limiting single failure for each event...*”

Response to Question 15.06.01-2:

A response to this question will be provided by October 30, 2008.

Question 15.06.01-3:

This event as presented in Section 15.6.1 shows an initial power increase due to boron reactivity feedback. Explain how the boron feedback as well as axial and radial power profiles are treated conservatively.

Regulatory basis: SRP 15.6.1, p 5, *“The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution”*

Response to Question 15.06.01-3:

A response to this question will be provided by October 30, 2008.

Question 15.06.03-1:

Please clarify the first sentence in section 15.6.3.1, "...*the reactor trips automatically on low PZR pressure, high SG pressure, or **high PZR pressure.***" (emphasis added)

Response to Question 15.06.03-1:

A response to this question will be provided by October 30, 2008.

Question 15.06.03-2:

Please present the results of the analyses that led to the conclusion that the case presented is the limiting case for radiological release.

Response to Question 15.06.03-2:

A response to this question will be provided by October 30, 2008.

Question 15.06.03-3:

Please present the results of the SG overfill analyses.

Regulatory basis: SRP 15.0, p 10, "*The reviewer ensures that the applicant has presented the results of analyses...*"

Response to Question 15.06.03-3:

A response to this question will be provided by October 30, 2008.

Question 15.06.03-4:

Please clarify where the SG apex is located (Fig 15.6-26).

Response to Question 15.06.03-4:

A response to this question will be provided by September 24, 2008.

U.S. EPR Final Safety Analysis Report Markups

Table 15.1-1—Decrease in Feedwater Temperature - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW _t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1113 psia
MFW flow rate	1444 lb _m /s
MFW temperature	446°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
Fuel-to-cladding gap conductance	13,445.20 BTU/(hr-ft ² -°F)
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm pcm/°F

15.01.01 - 15.01.04-7



I

Table 15.1-4—Increase in Feedwater Flow - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW _t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1176 psia
Main steam/feed flow rate	1267 lb _m /s
MFW temperature	346°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm pcm/°F

15.01.01 - 15.01.04-7



Table 15.1-7—Increase in Steam Flow - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW _t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1176 psia
Main steam/feed flow rate	1267 lb _m /s
MFW temperature	346°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
Fuel-to-cladding gap conductance	13445.20 BTU/(hr-ft ² -°F)
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm pcm/°F

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Table 15.1-10—Inadvertent Opening of an SG Relief or Safety Valve - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW _t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1176 psia
Main steam/feed flow rate	1267 lb _m /s
MFW temperature	346°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
Fuel-to-cladding gap conductance	13445.20 BTU/(hr-ft ² -°F)
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm pcm /°F

Note:

1. Parameter is nominal for the decreased MFW temperature value.

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