Response to

Request for Additional Information No. 34 Supplement 1, 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 15-8:

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:

The evaluation of the incore monitoring system is conducted as follows:

A new U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9 is added to provide an overview of the incore transient methodology.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9 will be added as described in the response and indicated on the enclosed markup.

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:

Representative plots of the minimum departure from nucleate boiling ratio (DNBR) and maximum linear power density (LPD) to their respective specified acceptable fuel design limits (SAFDLs) are shown on new U.S. EPR FSAR, Tier 2, Figures 15.1-57 through Figure 15.1-60. U.S. EPR FSAR, Tier 2, will also be revised to incorporate the description of these figures.

In the new figures, the line indicated as LPD is the calculated LPD normalized to either the fuel centerline melt (FCM) or clad strain limit (whichever is more limiting), including all applicable uncertainties. For anticipated operational occurrence (AOO) events, the limiting LPD of clad strain is more limiting and is therefore used to provide the normalized LPD. The LPD SAFDL is designed to protect both the FCM and clad strain limits as discussed in U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9.

The departure from nucleate boiling (DNB) reactor trip (RT) and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to initiate an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB limiting condition for operation (LCO) and LPD LCO are adequate to protect the SAFDL. This demonstrates that the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The following figures are provided in the enclosed U.S. EPR FSAR, Tier 2, markup:

- Figure 15.1-57—Decrease in Feedwater Temperature – Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL
- Figure 15.1-58—Increase in Main Feedwater Flow – Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL
- Figure 15.1-59—Increase in Steam Flow – Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL
- Figure 15.1-60—Inadvertent Opening of a SG Relief or Safety Valve – Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL

The subsections of the U.S. EPR FSAR, Tier 2, Section 15.1 are shown on the enclosed markup to clarify that both the low DNBR channel algorithm and the high LPD channel algorithm, as described in U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9, are used in this analysis.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.1.1.2, Section 15.1.1.3, Section 15.1.1.5, Section 15.1.2.2, Section 15.1.2.3, Section 15.1.2.5, Section 15.1.3.1, Section 15.1.3.2, Section 15.1.3.3, Section 15.1.3.5, Section 15.1.4.1, Section 15.1.4.2, Section 15.1.4.3.1, and Section 15.1.4.5 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR FSAR, Tier 2, Figure 15.1-57, Figure 15.1-58, Figure 15.1-59, and Figure 15.1-60 will be added 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:

For the increase in feedwater flow event, the main feedwater (MFW) flow is instantaneously increased to a conservative value to a single steam generator (SG). Total nominal MFW flow is 20,866,894 lb_m/hr or 1449.1 lb_m/sec per SG. Maximum MFW flow with valves wide open is 21,868,494 lb_m/hr or 1518.6 lb_m/sec per SG. This means that the MFW system is capable of supplying an additional 278 lb_m/sec (4 x (1518.6 -1449.1)) of flow above the nominal value. If all of the additional capacity is assumed to go to the faulted SG, the total flow is 1727.1 lb_m/sec (1449.1 lb_m/sec + 278 lb_m/sec).

The MFW density at nominal conditions is about 52 lb_m/ft^3. The MFW density at a reduced MFW temperature of 346˚F is about 56 lb_m/ft^3. Adjusting the 1727.1 lb_m/sec value by the density ratio yields a flow of 1860 lb_m/sec at the reduced temperature, which is rounded up to 1900 lb_m/sec.

Using a MFW flow rate of 1267 lb_m/sec calculated at hot full power (HFP) and a reduced MFW temperature of 346˚F steady-state conditions, the percentage increase in MFW flow is: (1900/1267) lb_m/sec x 100 percent = 150 percent. A 50 percent increase in feedwater flow to a single SG is conservative, because the valves wide open flow rate only allows for a 5 percent increase in feedwater flow above the nominal full power flow rate.

Given the conservative 50 percent increase in feedwater flow to one SG in conjunction with a 100˚F decrease in feedwater temperature to all four SGs, an additional reduction in feedwater flow to the unaffected SGs, in order to increase the severity of the event, is not considered necessary. U.S. EPR FSAR, Tier 2, Section 15.1.2.2 will be revised to include the 100˚F temperature reduction assumption.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.1.2.2 will be revised as described in the response and indicated on the enclosed markup.

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:

The initiating event for U.S. EPR FSAR, Tier 2, Section 15.1.4, "Inadvertent Opening of a Steam Generator Relief or Safety Valve," is an inadvertent opening of the steam generator (SG) loop 3 main steam relief isolation valve (MSRIV) (affected SG) in conjunction with a single failure of the associated main steam relief control valve (MSRCV) to close (see U.S. EPR FSAR, Tier 2, Table 15.1-12—Inadvertent Opening of an SG Relief or Safety Valve - Sequence of Events). The MSRCV on SG-3 (instead of SG-4) fails to close rather than fails to open as the initial condition of the MSRCV at 100 percent power is normally full open.

The analyses of initiating events with a symmetric core configuration (e.g., U.S. EPR FSAR, Tier 2, Sections 15.1.1, 15.1.3, and 15.1.4) are simulated as occurring in SG loop 3. The analyses of initiating events with asymmetric core configurations (e.g., U.S. EPR FSAR, Tier 2, Sections 15.1.2, and 15.1.5) are simulated as occurring in SG loop 4. Changes will be made to U.S. EPR FSAR, Tier 2, Table 15.1-11—Inadvertent Opening of an SG Relief or Safety Valve – Key Equipment Status to clarify that the MSRCV failure to close is on SG loop 3.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 15.1-11 will be revised as described in the response and indicated on the enclosed markup.

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:

The slightly higher cold leg response of loop 3, shown in U.S. EPR FSAR, Tier 2, Figure 15.1-37—Inadvertent Opening of a SG Relief or Safety Valve – Cold Leg Temperatures, is a result of the pressurizer (PZR) being located on this loop and the loss of mixing after the reactor coolant pumps (RCPs) trip. As the reactor coolant system (RCS) volume shrinks due to the temperature decrease (due to the decrease in feedwater temperature) there is an out surge of higher temperature water from the PZR between 20 seconds and 50 seconds as shown in U.S. EPR FSAR, Tier 2, Figure 15.1-41—Inadvertent Opening of a SG Relief or Safety Valve - Pressurizer Level. 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) slightly elevated above the other three loops. The heat transfer rate across the three unaffected SGs (i.e., SG-1, SG-2, and SG-4) is equal (as shown in U.S. EPR FSAR, Tier 2 Figure 15.1-36—Inadvertent Opening of a SG Relief or Safety Valve – Heat Transfer to SGs). SG 3 has a slightly higher heat transfer rate between 30 and 80 seconds, which is related to the higher SG inlet temperature.

FSAR Impact:

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

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:

For the hot zero power (HZP) cases, the analysis concludes that the minimum departure from nucleate boiling ratio (DNBR) is 2.77 and the peak linear power density (LPD) for this event is 12.02 kW/ft. The maximum LPD realized for this event is significantly below the LPD limit (17.2 kW/ft as shown in U.S. EPR FSAR, Tier 2, Table 4.1-1—Summary of U.S. EPR Reactor Design and Performance Characteristics) that precludes both one percent clad strain and fuel centerline melt (FCM). This supports the conclusion in Section 15.1.4.3.1 that the peak return to power does not cause fuel damage.

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 spectrum of cases, including break size and location, power level, offsite power status, and single failure, are performed for the main steam line break (MSLB) event. The results from this spectrum are compiled in Table 15.01.05-2-1—Summary of MSLB (Post-Scram) Analyzed Events.

FSAR Impact:

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

Table 15.01.05-2-1—Summary of MSLB (Post-Scram) Analyzed Events (3 Sheets)

Initial Condition	Break Size (ft^2)	Break Location Relative to MSIV	Single Active Failure	Offsite Power Status	Peak Post-Scram Reactor Power (MW_t)	Core Design	Cycle	Peak LHGR (kW/ft)	FCM Fuel Failure (%)	Fuel Failure at Clad Strain Limit (%)
HFP	4.12[a]	Upstream	MSRCV	Available	598.24	18-mo 18-mo	1 3	18.2 19.5	0.00 0.00	0.41 1.24
HFP	4.12	Upstream	MSRCV	Lost	255.27	18-mo 18-mo	1 3	8.3 4.8	0.00 0.00	0.00 0.00
HFP	1.71[b]	Upstream	MSRCV	Available	649.28	18-mo 18-mo	1 3	18.8 20.2	0.00 0.00	0.83 1.24
HFP	1.71	Upstream	MSRCV	Lost	266.16	18-mo 18-mo	1 3	9.0 5.2	0.00 0.00	0.00 0.00
HFP	1.71	Downstream	MSIV	Available	634.82	18-mo 18-mo	1 3	18.6 19.9	0.00 0.00	0.41 1.24
HFP	1.06[c]	Upstream	MSRCV	Available	492.31	18-mo 18-mo	1 3	16.8 18.0	0.00 0.00	0.00 0.41
HFP	1.06	Upstream	MSRCV	Lost	231.80	18-mo 18-mo	1 3	7.2 4.0	0.00 0.00	0.00 0.00
60% Power	4.12	Upstream	MSRCV	Available	564.88	18-mo 18-mo	1 3	18.3 19.5	0.00 0.00	0.41 1.24
60% Power	4.12	Upstream	MSRCV	Lost	237.72	18-mo 18-mo	1 3	7.0 3.8	0.00 0.00	0.00 0.00
60% Power	1.76[d]	Upstream	MSRCV	Available	626.16	18-mo 18-mo	1 3	19.1 20.3	0.00 0.00	0.83 1.24
60% Power	1.76	Upstream	MSRCV	Lost	252.34	18-mo 18-mo	1 3	7.6 4.2	0.00 0.00	0.00 0.00
60% Power	1.06	Upstream	MSRCV	Available	458.72	18-mo 18-mo	1 3	16.7 17.8	0.00 0.00	0.00 0.41
60% Power	1.06	Upstream	MSRCV	Lost	212.74	18-mo 18-mo	1 3	5.9 3.4	0.00 0.00	0.00 0.00
25% Power	4.12	Upstream	MSRCV	Available	470.15	18-mo 18-mo	1 3	18.0 18.8	0.00 0.00	0.41 1.24
25% Power	4.12	Upstream	MSRCV	Lost	193.10	18-mo 18-mo	1 3	3.6 3.8	0.00 0.00	0.00 0.00
25% Power	1.76[e]	Upstream	MSRCV	Available	516.46	18-mo 18-mo	1 3	19.0 19.9	0.00 0.00	0.83 1.24

Table 15.01.05-2-1—Summary of MSLB (Post-Scram) Analyzed Events (3 Sheets)

Initial Condition	Break Size (ft^2)	Break Location Relative to MSIV	Single Active Failure	Offsite Power Status	Peak Post-Scram Reactor Power (MW_t)	Core Design	Cycle	Peak LHGR (kW/ft)	FCM Fuel Failure (%)	Fuel Failure at Clad Strain Limit (%)
25% Power	1.76	Upstream	MSRCV	Lost	193.19	18-mo 18-mo	1 3	3.6 3.8	0.00 0.00	0.00 0.00
25% Power	1.06	Upstream	MSRCV	Available	387.61	18-mo 18-mo	1 3	16.3 17.0	0.00 0.00	0.00 0.00
25% Power	1.06	Upstream	MSRCV	Lost	154.64	18-mo 18-mo	1 3	3.2 3.4	0.00 0.00	0.00 0.00
HZP	4.12	Upstream	MSRCV	Available	996.89	18-mo 18-mo	1 3	17.2 18.8	0.00 0.00	0.41 1.24
HZP	4.12	Upstream	MSRCV	Lost	376.05	18-mo 18-mo	1 3	15.9 11.6	0.00 0.00	0.00 0.00
HZP	1.72[f]	Upstream	MSRCV	Available	1062.3[g]	18-mo 18-mo	1 3	17.5 19.1	0.00 0.00	0.41 1.24
HZP	1.72	Upstream	MSRCV	Lost	394.3	18-mo 18-mo	1 3	16.7 12.7	0.00 0.00	0.00 0.00
HZP	1.72	Downstream	MSIV	Available	1039.3	18-mo 18-mo	1 3	17.4 19.0	0.00 0.00	0.41 1.24
HZP	1.72	Upstream	None	Available	1062.2	18-mo 18-mo	1 3	17.5 19.1	0.00 0.00	0.41 1.24
HZP	1.06	Upstream	MSRCV	Available	817.14	18-mo 18-mo	1 3	16.3 18.0	0.00 0.00	0.00 1.24
HZP	1.06	Upstream	MSRCV	Lost	343.29	18-mo 18-mo	1 3	14.2 10.1	0.00 0.00	0.00 0.00
HZP P12	0.12[h]	Downstream	PS Channel	Available	135.01	18-mo 18-mo	1 3	5.5 5.2	0.00 0.00	0.00 0.00

Notes:

a) This is the size of a double-ended guillotine break.

b) This is the break size that results in the greatest post-scram reactor power for cases initiated from full-power conditions.

c) This is the smallest break size that results in a reactor trip for all cases initiated from full-power or part-power conditions.

d) This is the break size that results in the greatest post-scram reactor power for cases initiated from 60%-power conditions.

e) This is the break size that results in the greatest post-scram reactor power for cases initiated from 25%-power conditions.

f) This is the break size that results in the greatest return to power for cases initiated from hot-zero-power conditions.

g) The return to power for this failed-open-MSRCV case is not significantly greater than that for the corresponding no-active-failure case, and its fuel failure results are identical. However, since its radiological consequences (with prolonged steam release through the failed open MSRCV) are significantly worse, its single active failure has been assumed for the other cases in the matrix of initial power levels and break sizes.

h) This is the largest break size at hot-zero-power conditions that, with primary system pressure less than the P12 permissive setpoint and offsite power remaining available, does not result in automatic MSIV closure and MFW isolation (because Low SG Pressure actuation of these functions is disabled below the P12 setpoint).

Question 15.01.05-3:

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

Response to Question 15.01.05-3:

For the pre-scram main steam line break (MSLB), the potential exists for a delay in reactor trip (RT) because power decalibration and harsh containment conditions potentially challenge specified acceptable fuel design limits (SAFDL) (i.e., cladding strain and/or fuel centerline melt (FCM)).

Power decalibration is caused by reactor vessel (RV) downcomer fluid density changes, which result in power-range ex-core detector shadowing during heatup or cooldown transients. The nuclear power level indicated by the ex-core detectors is lower than the actual reactor power level when the coolant entering the RV is cooler than the normal full-power temperature (and higher when the inlet coolant is warmer than the normal full-power temperature). This effect is taken into account in the modeling of power-dependent RTs credited in the pre-scram MSLB analysis.

The release of steam can cause harsh environmental conditions. Under such conditions, only those trips which have been qualified for harsh environments are credited, and increased uncertainties are included in all environmentally qualified trip setpoints.

Although this event is usually dispositioned for existing pressurized water reactor (PWR) designs as being bounded by the post-scram MSLB events, it is evaluated for the U.S. EPR based on the potential impact of the anticipated harsh environmental conditions in the sensor area of the various steam line break locations and because of the asymmetric core effects of the post-scram MSLB event versus the anticipated symmetric effects of the pre-scram MSLB events. A pre-scram MSLB event is considered symmetric because of the absence of main steam line (MSL) check valves in the U.S. EPR design and the presence of a main steam (MS) bypass common header upstream of the turbine stop valves.

The analysis of the pre-scram portion of the MSLB uses the NRC approved methodology described in EMF-2310(P)(A), “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors” and ANP-10263(P)(A), “Codes and Methods Applicability Report for the U.S. EPR.” The methodology uses the S-RELAP5 version of the RELAP code to model PWR system thermal-hydraulic response.

Pre-scram MSLB cases are chosen to evaluate the effects of core power, loss of offsite power (LOOP), automatic rod control (ARC), break size, and break location. This set of cases is performed for various core power levels (hot full power (HFP), 60 percent, 25 percent, and hot zero power (HZP)) as well as for each break location. The break locations are:

- Downstream break location – includes main steam isolation valve (MSIV) failure on the broken steam line, break location downstream of the failed MSIV, and harsh Safeguard Building conditions.
- Upstream break location outside containment – includes a safety injection system (SIS) train failure on the reactor coolant system (RCS) loop with the broken steam line, break location upstream of the MSIV, and harsh Safeguard Building conditions.

- Upstream break location inside containment – includes a SIS train failure on the RCS loop with the broken steam line, break at the steam generator (SG) outlet, and harsh containment environmental conditions.

To support the small break, pre-scram evaluation of the MSLB event, a representative plot of the normalized minimum departure from nucleate boiling ratio (DNBR) and maximum linear power density (LPD) to their respective SAFDL is provided in the enclosed new U.S. EPR FSAR, Tier 2, Figure 15.1-61—MSLB (small break, pre-scram) – Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL.

The line indicated as LPD is the calculated LPD normalized to either the fuel centerline melt (FCM) or clad strain limits (whichever is more limiting), including all applicable uncertainties. For anticipated operational occurrence (AOO) events, the limiting LPD of clad strain is more limiting and is therefore used to provide the normalized LPD. The LPD SAFDL is designed to protect both the FCM and clad strain limits as discussed in U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9. This applies only to the small break, pre-scram portion of the MSLB event.

The departure from nucleate boiling (DNB) RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB limiting condition for operation (LCO) and LPD LCO are adequate to protect the SAFDL. This demonstrates that for the small break, pre-scram portion of the MSLB event, the fuel cladding integrity is maintained and peak centerline temperatures remain below the FCM limit.

Medium-break and double-ended guillotine break analyses are performed with a deterministic LYNXT code calculation. The results show that the pre-scram, at power cases have a minimum DNBR approximately equal to the DNBR LCO and a maximum LPD of 14.99 kW/ft. Therefore, the pre-scram portion of the MSLB event does not challenge the DNB or LPD SAFDLs.

U.S. EPR FSAR, Tier 2, Section 15.1.5 will be clarified and new Figure 15.1-61 will be added.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.1.5.2 and Section 15.1.5.3 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR FSAR, Tier 2, Figure 15.1-61 will be added as described in the response and indicated on the enclosed markup.

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:

This represents a typographical error, which should read "unaffected SGs." U.S. EPR FSAR, Tier 2, Section 15.1.5.1 will be corrected.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.1.5.1 will be revised as described in the response and indicated on the enclosed markup.

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:

The 649.28 MWt value is the hot full power (HFP) peak return to power. The hot zero power (HZP) peak return to power is 1062.3 MWt (23.14 percent rated thermal power (RTP)). U.S. EPR FSAR, Tier 2, Table 15.1-16—MSLB - Calculated Fuel Parameters inadvertently contains the HFP information. U.S. EPR FSAR, Tier 2, Table 15.1-16 will be revised to include the correct value.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 15.1-16 will be revised as described in the response and indicated on the enclosed markup.

Question 15.01.05-12:

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

Response to Question 15.01.05-12:

As shown in U.S. EPR FSAR, Tier 2, Figure 15.1-55—MSLB - Reactor Power, there is a sudden spike in reactor power just prior to the end of the return to power. This power increase is a consequence of a chain of events initiated by the drying out of the affected steam generator (SG). As the affected SG begins to dry out, the choked flow condition at the SG outlet nozzle terminates and SG pressure falls rapidly, enhancing boiling heat transfer on the tubes.

While the SG U-tubes are modeled by means of twelve axial nodes, two of which represent the tube sheet region and ten heat conductors that provide heat transfer to the SG boiler region, the boiler region is modeled with a single node. In the physical SG, the small affected SG water inventory during the return to power would be in contact only with the tube surfaces near the tube sheet. In the modeling of the boiler region, all SG tube thermal conductors are in contact with the boiler region water inventory.

During the return to power, the great majority of the heat transfer from the reactor coolant in the tubes to the affected SG boiler region is via the first tube thermal conductor. When the SG begins to experience dryout, this heat conductor experiences departure from nucleate boiling (DNB), and the majority of heat transitions to the next tube heat conductor, which soon also experiences DNB. This initiates a cascading chain of events that ends when the last tube heat conductor experiences DNB.

During the DNB cascade, net SG heat transfer falls off rapidly. Prior to its experiencing DNB, the mass of reactor coolant adjacent to tube conductor #1 passes through tube conductor #2 and all subsequent tube conductors prior to their experiencing DNB. Hence, in spite of the net reduction in SG heat transfer, a slug of reactor coolant experiences enhanced heat transfer throughout its transit through the affected SG tubes and is overcooled. When this overcooled slug of water reaches the core, it adds positive reactivity and causes the spike in power.

While this power spike is caused by a brief enhancement of SG tube heat transfer associated with dryout, its magnitude is an artifact of the single-node boiler region in the main steam line break (MSLB) model. As much of the cooling of the reactor coolant occurs in tube regions that would be above the water level in the steam generators, the single-node boiler region overpredicts the cooling of the reactor coolant and the resultant power spike.

FSAR Impact:

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

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:

The effects of loss of offsite power (LOOP) and no-LOOP on the provided cases are as follows:

- Assuming LOOP coincident with the reactor trip results in a coast down of the reactor coolant pumps (RCPs) during the early portion of the turbine trip (TT) event. The reduction in reactor coolant system (RCS) flow results in a decrease in the heat transfer from the RCS to the secondary system. The reduction in heat transfer to the secondary system exacerbates the heatup and pressurization of the RCS. Therefore, assumption of LOOP is conservative for the TT peak RCS pressure case.
- Conversely, assuming no-LOOP is conservative for the TT peak secondary pressure case. Continued operation of the RCPs maximizes the energy transfer from the RCS to the secondary side. The TT peak secondary pressure case is bounded by the results of the main steam isolation valve closure (MSIVC) event.

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:

From a departure from nucleate boiling ratio (DNBR) perspective, the loss of external load (LOEL), turbine trip (TT), and loss of condenser vacuum (U.S. EPR FSAR, Tier 2, Sections 15.2.1, 15.2.2, and 15.2.3, respectively) events are similar. The initiating event causes an immediate closure of the turbine control valve or turbine stop valve, which results in a sudden increase in pressure and gradual increase in core inlet temperature.

The TT event is used to provide an evaluation of DNBR for these events. Because of the fast closure of the turbine stop valve, there is neither increase in the reactor coolant pump (RCP) speed nor flow that occurs in the LOEL event due to the temporary increase in turbine generator (TG) speed.

For the TT transient DNBR analysis, a qualitative evaluation is performed of the key parameters affecting DNBR up to the point of rod insertion due to a reactor trip (RT) on high pressurizer (PZR) pressure. The evaluation of these key parameters (including core inlet temperature, core exit pressure, core inlet mass flux, and fuel rod surface heat flux) concludes that the TT event is not a DNBR-limiting event. To support this conclusion, the key parameters affecting DNBR are shown in Figure 15.02.01-15.02.05-4-1—Turbine Trip – Representative Plot of the Key Parameters to DNBR Normalized to Initial Value. Each parameter has been normalized to its initial value so that the behavior of relevant parameters can be displayed in a single graph to provide a perspective on the magnitude of change that occurs in each parameter.

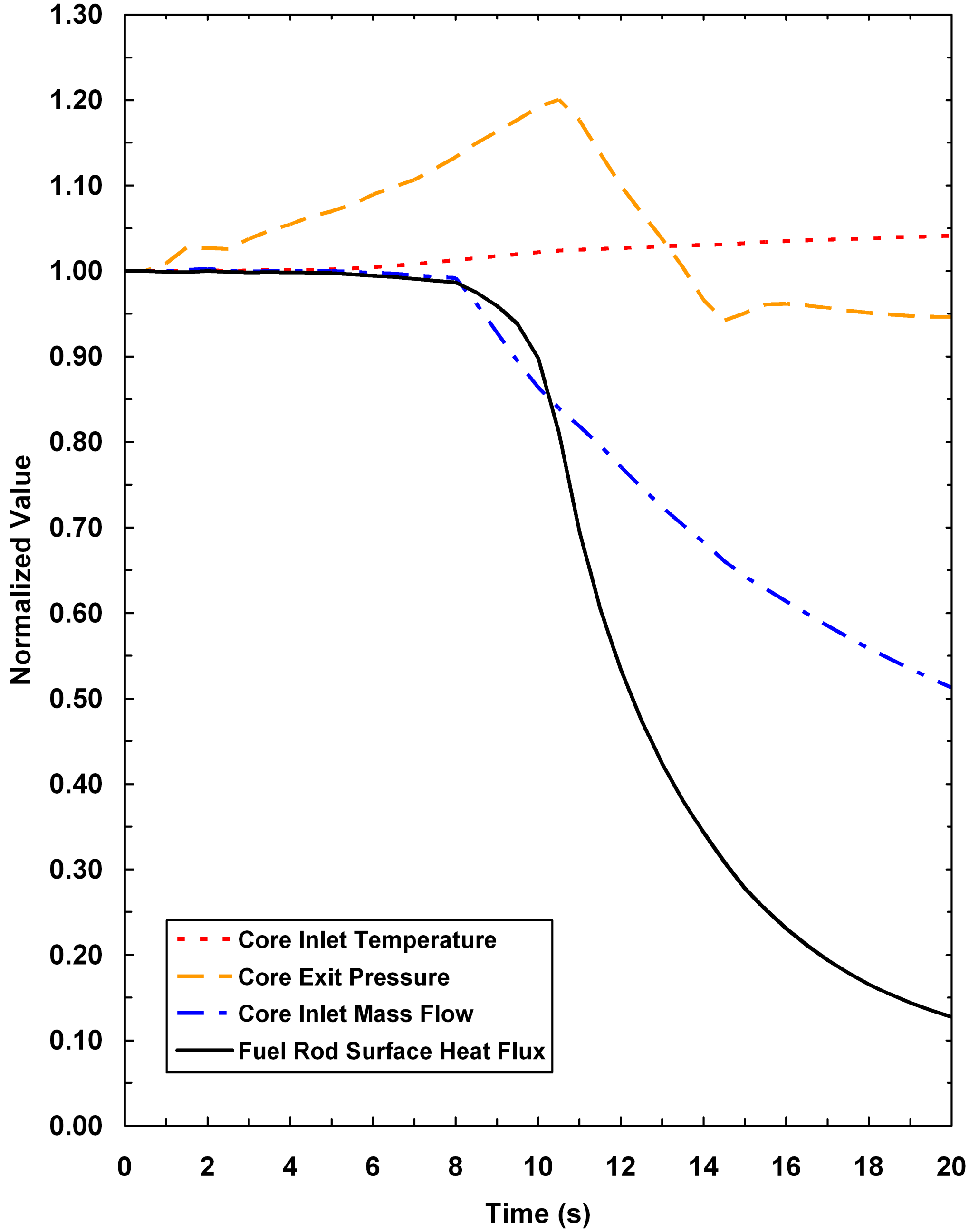
Examination of the transient data reveals that the maximum heat flux occurs at 0.00 seconds. U.S. EPR FSAR, Tier 2, Table 15.2-3—Turbine Trip-RCS Overpressurization - Sequence of Events shows that rod insertion occurs at approximately 7.4 seconds. At 0.00 seconds, the minimum DNBR is restricted by the DNB limiting conditions for operation (LCO). At 7.4 seconds, the steady mass flow (99.4 percent of initial), large increase in pressure (268 psi increase), and decreased heat flux (98.9 percent of initial) will offset the degradation due to the increased temperature (5.55°F increase). The DNBR will not degrade in comparison to the initial condition of the transient. Therefore, the event's minimum DNBR is the DNB LCO.

A representative plot of the minimum DNBR as a function of time for the closure of a main steam isolation valve (MSIV) for U.S. EPR FSAR, Tier 2, Section 15.2.4 is provided in the response to Question 15.02.01-15.02.05-6.

FSAR Impact:

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

Figure 15.02.01-15.02.05-4-1—Turbine Trip – Representative Plot of the Key Parameters to DNBR Normalized to Initial Value



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.

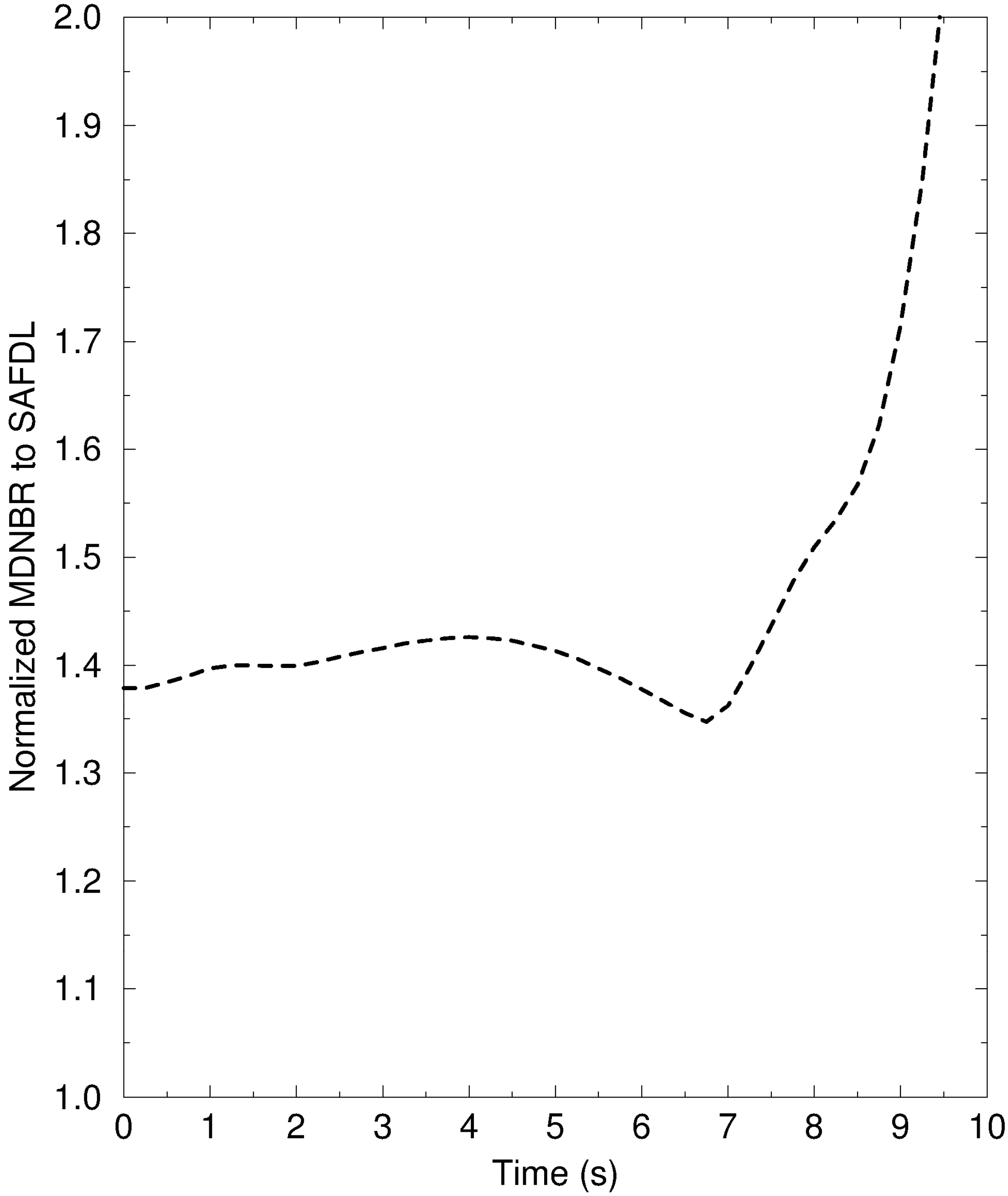
Response to Question 15.02.01-15.02.05-6:

To support the conclusion in the U.S. EPR FSAR, Tier 2, Section 15.2.4.3.2, a representative minimum departure from nucleate boiling ratio (DNBR) as a function of time normalized to the DNBR specified acceptable fuel design limit (SAFDL) calculated by the low DNBR channel algorithm is provided in Figure 15.02.01-15.02.05-6-1—Single Main Steam Isolation Valve Closure – Representative Plot of Minimum DNBR Normalized to the DNBR SAFDL. This transient event is not limiting with respect to departure from nucleate boiling (DNB) or linear power density (LPD) and does not incur an incore trip.

FSAR Impact:

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

Figure 15.02.01-15.02.05-6-1—Single Main Steam Isolation Valve Closure – Representative Plot of Minimum DNBR Normalized to the DNBR SAFDL



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:

U.S. EPR FSAR, Tier 2, Section 15.2.4.3.2 provides an overview of the S-RELAP5 modeling for the main steam isolation valve (MSIV) closure event.

To provide an evaluation of the minimum departure from nucleate boiling ratio (DNBR) for the affected region of the core, the affected loop's temperature is applied; the affected loop's temporal response is adjusted to account for loop transport time (from sensor to affected core region). To conservatively model the power response, the fuel rod surface heat flux is assumed constant up to the point of rod insertion due to the reactor trip (RT) on high steam generator (SG) pressure. The core inlet mass flow and core exit pressure are provided from S-RELAP5.

A deterministic DNBR evaluation is performed using the LYNXT code at the starting condition and point of rod insertion. The DNBR degradation calculated with LYNXT is near negligible (-0.03) from the initial condition, resulting in a conservatively calculated minimum DNBR of 2.08.

To provide a representative transient plot of minimum DNBR (see the response to question 15.02.01-15.02.05-6) the same adjustments to the boundary conditions described above are applied and evaluated with the Low DNBR Channel algorithm; the only difference here is that the starting condition is the DNB limiting condition for operation (LCO).

For clarification, U.S. EPR FSAR, Tier 2, Section 15.2.4.2 will be revised to state that the algorithm described in U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9 is the primary analysis tool for this analysis; however, a deterministic evaluation using LYNXT is also performed.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.2.4.2 will be revised as described in the response and indicated on the enclosed markup.

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:

The analysis described in U.S. EPR FSAR, Tier 2, Section 15.2.4.3.2 uses cold leg temperatures in combination with point kinetics to determine reactivity in the RELAP5 model. This combination results in a slight increase in the reactor power and a more conservative analysis. U.S. EPR FSAR, Tier 2, Section 15.2.4.3.2 will be revised to remove the statement concerning a 3-D analysis.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.2.4.3.2 will be revised as described in the response and indicated on the enclosed markup.

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:

The loss of non-emergency power (LNEP), described in U.S. EPR FSAR, Tier 2, Section 15.2.6, is defined as the loss of offsite power (LOOP) or loss of non-emergency AC power to the station auxiliaries. It is initiated by either a complete LOOP coincident with a turbine trip (TT), or a loss of onsite power. Diesel generators are started and provide electric power to vital loads. The sensible and decay heat loads are handled by the steam relief and emergency feedwater (EFW) systems. Different segments of this event have similarities to the complete loss of normal feedwater flow (LNFF) and complete loss of reactor coolant system (RCS) flow events.

U.S. EPR FSAR, Tier 2, Section 15.2.6 will be revised to describe the sequence of events that follow a loss of non-emergency power.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.2.6 will be revised as described in the response and indicated on the enclosed markup.

Question 15.02.07-1:

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

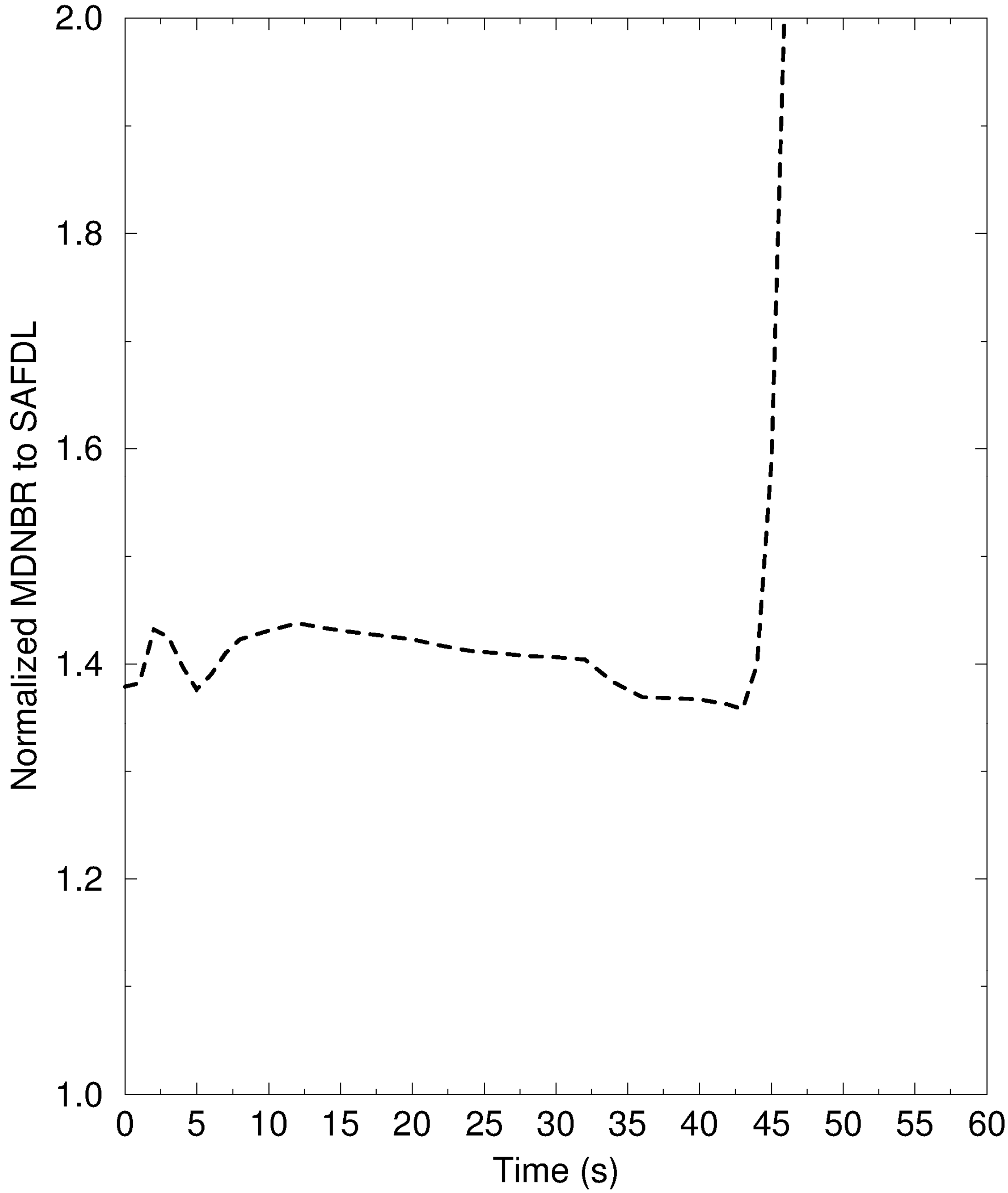
Response to Question 15.02.07-1:

A representative minimum departure from nucleate boiling ratio (DNBR) as a function of time normalized to the DNBR specified acceptable fuel design limit (SAFDL) calculated by the low DNBR channel algorithm is provided in Figure 15.02.07-1-1—Loss of Normal Feedwater Flow – Representative Plot of Minimum DNBR Normalized to the DNBR SAFDL. This transient event is not departure from nucleate boiling (DNB) or linear power density (LPD) limiting and does not incur an incore trip.

FSAR Impact:

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

Figure 15.02.07-1-1—Loss of Normal Feedwater Flow – Representative Plot of Minimum DNBR Normalized to the DNBR SAFDL



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:

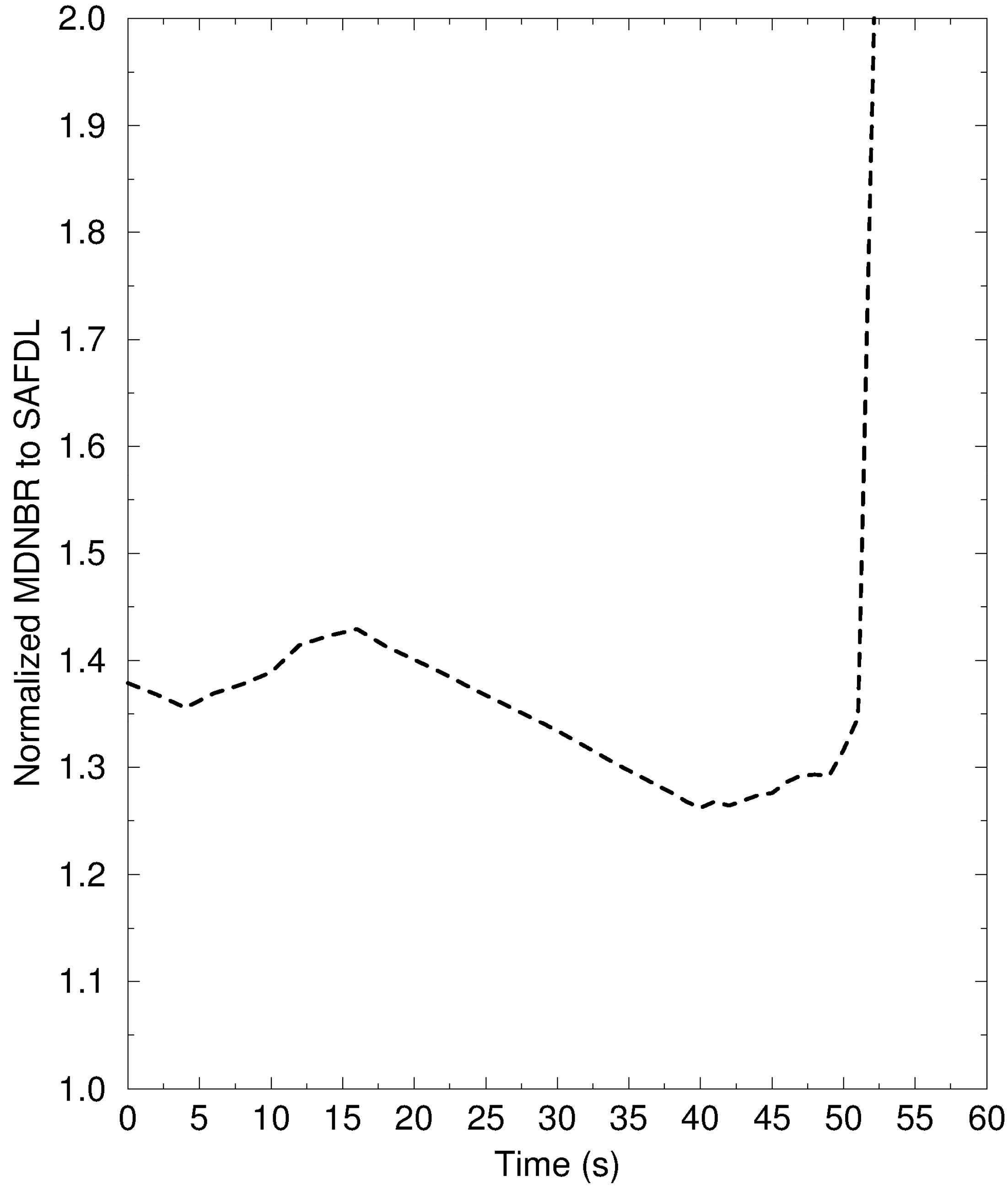
U.S. EPR FSAR, Tier 2, Section 15.2.8.1 states that the feedwater line break (FWLB) scenarios that cause a cooldown of the reactor coolant system (RCS) are bounded by the analysis described in U.S. EPR FSAR, Tier 2, Section 15.1.5. During the post-scram peak return to power with a stuck control rod for the U.S. EPR FSAR, Tier 2, Section 15.1.5 event, 1.24 percent of the fuel assemblies are predicted to fail by exceeding the clad strain limit. No fuel failures result from fuel melt. See the response to RAI Question 15.01.05-3 for a discussion of the pre-scram portion of this event.

To support the statements in the U.S. EPR FSAR, Tier 2, Section 15.2.8.2 for the scenarios that cause a cold leg heatup, a representative minimum departure from nucleate boiling ratio (DNBR) as a function of time normalized to the DNBR specified acceptable fuel design limit (SAFDL) calculated by the low DNBR channel algorithm is provided in Figure 15.02.08-2-1—Feedwater Piping Break Inside and Outside Containment – Representative Plot of Minimum DNBR Normalized to the DNBR SAFDL. This transient event is not departure from nucleate boiling (DNB) or linear power density (LPD) limiting and does not incur an incore trip.

FSAR Impact:

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

Figure 15.02.08-2-1—Feedwater Piping Break Inside and Outside Containment – Representative Plot of Minimum DNBR Normalized to the DNBR SAFDL.



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:

The U.S. EPR FSAR figures identified above and in RAI Questions 15.02.08-5, 15.02.08-6, 15.02.08-7, 15.02.08-8, and 15.02.08-10 are incorrect. U.S. EPR FSAR, Tier 2, Figure 15.2-65—FWLB Representative Small Break – RCS Maximum Pressure, Figure 15.2-66—FWLB Representative Small Break – Steam Generator Maximum Pressure, Figure 15.2-67, Figure 15.2-68, and Figure 15.2-69 will be replaced.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Figure 15.2-65 through Figure 15.2-69 will be revised as described in the response and indicated on the enclosed markup.

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:

This is addressed in the response to RAI Question 15.02.08-3.

FSAR Impact:

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

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:

This is addressed in the response to RAI Question 15.02.08-3.

FSAR Impact:

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

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:

This is addressed in the response to RAI Question 15.02.08-3.

FSAR Impact:

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

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:

This is addressed in the response to RAI Question 15.02.08-3.

FSAR Impact:

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

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:

This is addressed in the response to RAI Question 15.02.08-3.

FSAR Impact:

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

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:

The individual reactivity components used to compute the total reactivity parameters shown in U.S. EPR FSAR, Tier 2, Figure 15.2-5—Turbine Trip RCS Overpressurization - Total Reactivity and Figure 15.2-22—MSIVC Secondary Overpressurization—Total Reactivity are provided in Figure 15.02.08-14-1—Reactivity Components for Turbine Trip Peak RCS Pressure Case and Figure 15.02.08-14-2—Reactivity Components for MSIVC Peak Secondary Pressure Case, respectively.

FSAR Impact:

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

Figure 15.02.08-14-1—Reactivity Components for Turbine Trip Peak RCS Pressure Case

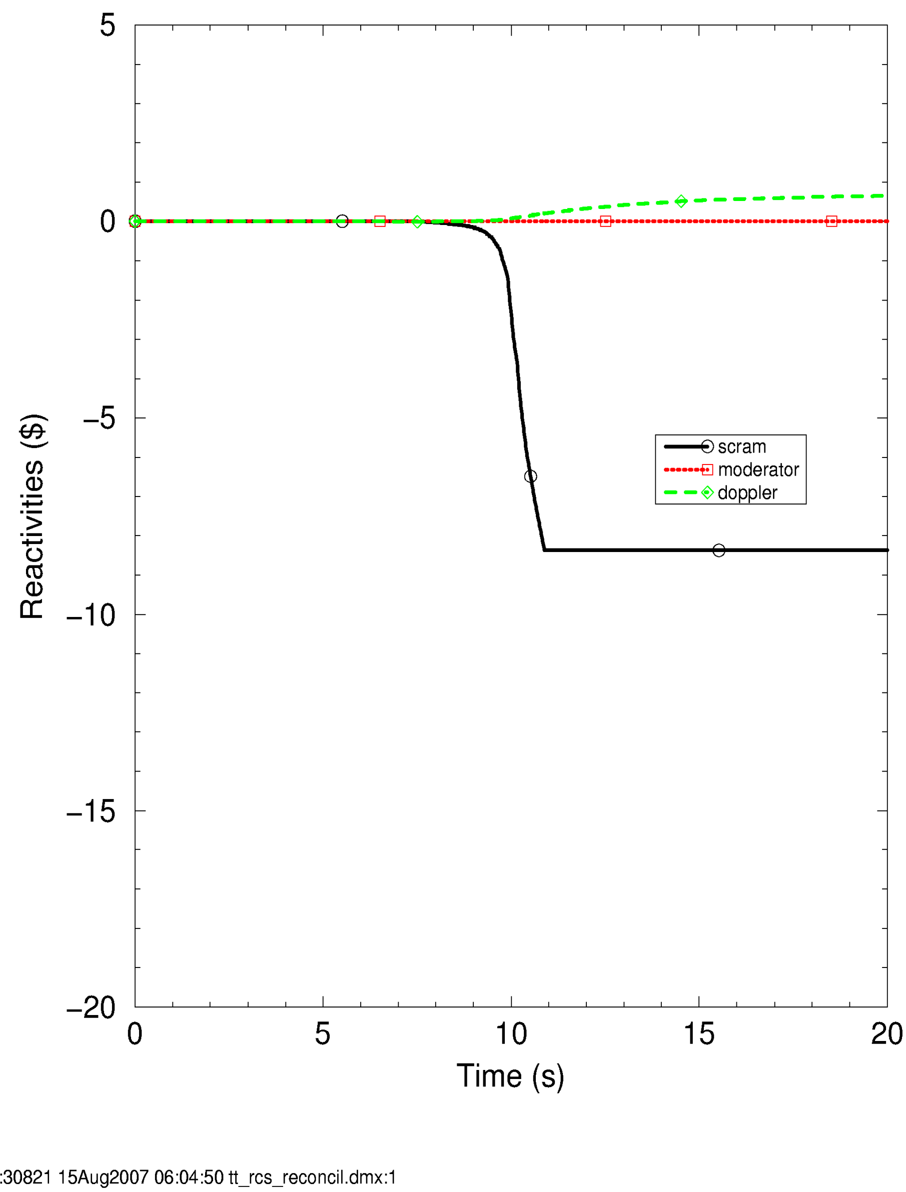
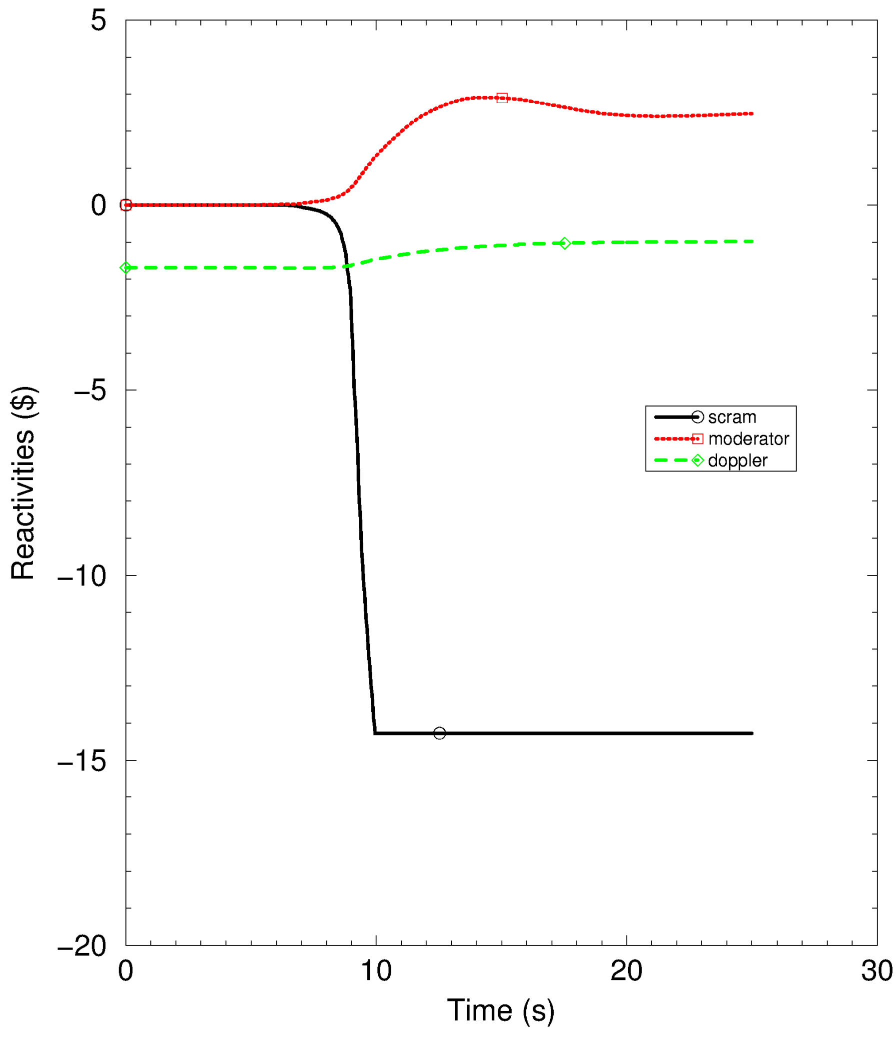


Figure 15.02.08-14-2—Reactivity Components for MSIVC Peak Secondary Pressure Case



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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 representative plot of the minimum departure from nucleate boiling ratio (DNBR) and maximum linear power density (LPD) to their respective specified acceptable fuel design limit (SAFDL) is shown in the enclosed new U.S. EPR FSAR, Tier 2, Figure 15.6-93 (IOPSRV – Representative Plot of Minimum Normalized Minimum DNBR and Maximum LPD Normalized to the SAFDL) for the inadvertent opening of a pressurizer safety relief valve (IOPSRV) event.

In the new figure, the line indicated as LPD is the calculated LPD, including all applicable uncertainties, normalized to either the fuel centerline melt (FCM) or clad strain limit (whichever is more limiting). For anticipated operational occurrence (AOO) events, the limiting LPD of clad strain is more limiting and is therefore used to provide the normalized LPD. The LPD SAFDL is designed to protect both the FCM and clad strain limits as discussed in U.S. EPR FSAR, Tier 2, Section 15.0.0.3.9.

The departure from nucleate boiling (DNB) reactor trip (RT) and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, the DNB limiting condition for operation (LCO) and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the FCM limit.

U.S. EPR FSAR, Tier 2, Section 15.6.1.2 and Section 15.6.1.3 will be revised to clarify the DNB and LPD trip functions.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 15.6.1.2 and Section 15.6.1.3 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR FSAR, Tier 2, Figure 15.6-93 will be added as described in the response and indicated on the enclosed markup.

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:

The boron feedback effect is modeled in the analysis via the relationship between the moderator temperature and density. Analyses are performed with the moderator temperature coefficient as the most positive at beginning-of-cycle or as the most negative at end-of-cycle to determine the conservative boron feedback effect.

The inadvertent opening of a pressurizer safety relief valve (IOPSRV) analysis presented in the U.S. EPR FSAR, Tier 2, Section 15.6.1 uses nominal axial and radial power profiles without additional conservatism. However, a sensitivity study was conducted utilizing the small break loss of coolant analysis (SBLOCA) methodology to evaluate the IOPSRV event with added conservatism. The core model in the SBLOCA method consists of a three-region radial representation of the core and considers a top-peaked axial power profile. In addition, the Moody critical flow model and a 20 percent multiplier on the decay heat are other conservative assumptions of the SBLOCA method. Conservative treatment of the break flow through the pressurizer safety relief valve (PSRV) and the reactor power decay heat is already considered in the reference analysis.

The results from the IOPSRV event analysis using the SBLOCA methodology predict no clad heat-up and calculate a hot channel core exit void fraction of about 0.6. The IOPSRV event analysis with the non-loss of coolant analysis (non-LOCA) methodology and with conservative PSRV flow and core decay heat assumptions produces results comparable to those obtained by using the SBLOCA methodology. Therefore, the penalizing assumptions of the SBLOCA analysis method do not significantly affect the response of the IOPSRV event.

FSAR Impact:

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

Question 15.06.03-1:

Please clarify the first sentence in section15.6.3.1, "*…the reactor trips automatically on low PZR pressure, high SG pressure, or **high PZR pressure.***" (emphasis added)

Response to Question 15.06.03-1:

The possibility of reactor trip (RT) on high pressurizer (PZR) pressure is based on the sequence of events observed in several cases during the analysis of the steam generator (SG) tube rupture event. The cases were initiated from a conservatively high PZR pressure of 2300 psia with a bounding beginning of cycle (BOC) moderator density reactivity coefficient. With no charging flow, the primary pressure decreases immediately after the event initiation and subsequently increases due to the reactor power increase as a result of positive moderator density reactivity feedback from the decreasing moderator density. The higher heat input from the reactor coolant system (RCS) results in a rise in SG pressure, causing the turbine control valves to close to maintain constant steam flow to the turbine. The closure of the turbine control valves results in a gradual increase in SG pressure. The increase in steam pressure results in an increase in RCS temperatures and PZR pressure until the reactor automatically trips on high PZR pressure.

FSAR Impact:

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

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:

The steam generator tube rupture (SGTR) thermal-hydraulic (TH) analysis was conducted in two phases. The initial phase consisted of running base hot and cold side break cases with no single failure at a number of potential initial conditions to determine a limiting initial condition for the second phase. The second phase analyzed various single failures. The initial condition selected for the single failure analyses assumes a reduced average reactor coolant temperature (Tavg) (584°F), 5 percent steam generator (SG) tube plugging, chemical volume control system (CVCS) charging pumps to be available, and loss of offsite power (LOOP) at reactor trip (RT).

Based on 25 TH cases analyzed, the radiological consequences were then evaluated for four single failure cases (Table 15.06.03-2-1—Potential Limiting Case Thermal Hydraulic Parameters) to determine the limiting radiological release case.

The radiological analyses were based on the following conditions:

a) One of the emergency feedwater (EFW) pumps is out of service for preventive maintenance (the affected SG in Cases 14a, 19 and 21, and one of the unaffected SGs in Case 15a).

b) A double-ended guillotine (DEG) break takes place at t=0 (a hot-side break at hot full power (HFP), 584°F T_{avg}).

c) The operator manually trips the reactor at 30 min in cases 14a, 15a, and 21, while there is an automatic trip at 940 sec (15.67 min) in Case 19 as a result of high pressurizer (PZR) pressure. In all cases, LOOP takes place upon RT.

d) With respect to the atmospheric releases via the affected SG, the scenarios are as follows:

Cases 14a and 15a
At 2400 sec (40 min) into the accident, the operator identifies the affected SG and resets (raises) its main steam relief train (MSRT) setpoint to terminate the atmospheric releases. A delay time of 1 min is conservatively assumed for valve closure (at 41 min).

Case 19
The affected SG MSRIV low steam pressure isolation setpoint (normal setpoint prior to being reset) is reached at 1556 sec (25.93 min), and MSRIV closure is initiated. A delay time of 1 min is conservatively assumed for valve closure (at 26.93 min).

Case 21
At 2400 sec (40 min) into the accident, the operator identifies the affected SG and resets its MSRT setpoint. The reset MSRIV low steam pressure setpoint is reached at 2570 sec (42.83 min), and MSRIV closure is initiated. A delay time of 1 min is conservatively assumed for valve closure (at 43.83 min).

e) Partial cooldown, at 90°F/hr via the three intact SG, is initiated by operator action at 2400 sec (40 min) for Cases 14a, 15a and 21, and automatically at 1194 sec (19.9 min) for Case 19.

f) Single Failure

The following conditions and system failures were considered in the analysis:

All Cases:	LOOP
Case 14a:	Main steam isolation valve (MSIV) of affected SG fails to close.
Case 15a:	EFW control valve of affected SG fails in the open position.
Case 19:	MSRCV of affected SG remains fully open (no CVCS).
Case 21:	Main steam relief control valve (MSRCV) of affected SG remains fully open (with CVCS).

<u>Source Term</u>

Two alternative SGTR source-term scenarios were postulated, as follows:

a) An SGTR with a pre-accident iodine spike, where a reactor transient had occurred prior to the postulated accident raising the primary coolant concentration to the proposed maximum value of 60 µCi/gm dose equivalent (DE) I-131.

b) An SGTR with an accident-induced concurrent iodine spike of 8-hour duration, where the iodine spike corresponds to an increase in the design-basis iodine appearance rate into the primary coolant by a factor of 335.

The two source-term scenarios are analyzed independently and the bounding case selected. Fuel damage is not postulated.

In accordance with the guidance in NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms" (3/7/2006), Section 9, the analysis considered the release of iodines, noble gases, and alkalis. All other radionuclides were assumed to remain within the liquid phase (reactor coolant system (RCS) and secondary coolant).

The post-SGTR submersion doses corresponding to the four thermal-hydraulic scenarios and the two iodine spikes are presented in Table 15.06.03-2-2—SGTR Submersion Doses. The following are noted:

a) The bounding dose is to the exclusion area boundary (EAB) for the concurrent iodine spike and the TH response data for Case 21, amounting to (0.731 rem / 2.5 rem) = 29 percent of the regulatory limit.

b) The worst-case dose to the main control room (MCR) is for the same scenario as for the EAB, and amounts to (0.60 rem / 5 rem) = 12 percent of the regulatory limit.

c) The MCR doses listed in Table 15.06.03-2-2 exclude direct shine from the filters. The latter amount to about 6 mrem for the accident duration.

FSAR Impact:

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

Table 15.06.03-2-1—Potential Limiting Case Thermal Hydraulic Parameters

Case	%SGTP (Steam Generator Tube Plugging)	Break Location	Average Temperature (°F)	Integrated Break Flow (lb_m)	Integrated Break Mass Flashed (lb_m)	Steam release from rupt'd SG (lb_m)	Steam release from intact SGs (lb_m)
Case14a	5	Hot Side	584	144000	9879	11854	1.65E6
Case15a	5	Hot Side	584	186000	10223	11452	1.44E6
Case 19	5	Hot Side	584	213000	6326	100027	1.64E6
Case 21	5	Hot Side	584	171000	10679	66950	1.52E6

Table 15.06.03-2-2—SGTR Submersion Doses

Spiking Option	Receptor	TEDE[a] Dose (rem)	Dose Limit (rem)
TH Response Case 14a			
Pre-accident iodine spike	EAB	8.948E-01	25
	LPZ[b]	2.517E-01	25
	MCR	2.412E-01	5
Concurrent iodine spike	EAB	5.530E-01	2.5
	LPZ	4.394E-01	2.5
	MCR	5.292E-01	5
TH Response Case 15a			
Pre-accident iodine spike	EAB	8.321E-01	25
	LPZ	2.434E-01	25
	MCR	2.266E-01	5
Concurrent iodine spike	EAB	5.244E-01	2.5
	LPZ	4.296E-01	2.5
	MCR	5.139E-01	5
TH Response Case 19			
Pre-accident iodine spike	EAB	8.294E-01	25
	LPZ	2.182E-01	25
	MCR	2.276E-01	5
Concurrent iodine spike	EAB	1.041E-01	2.5
	LPZ	3.920E-02	2.5
	MCR	3.716E-02	5
TH Response Case 21			
Pre-accident iodine spike	EAB	1.106E+00	25
	LPZ	2.947E-01	25
	MCR	2.899E-01	5
Concurrent iodine spike	EAB	7.308E-01	2.5
	LPZ	4.978E-01	2.5
	MCR	5.967E-01	5

Notes:

a) TEDE = Total effective dose equivalent.

b) LPZ = low population zone.

Question 15.06.03-4:

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

Response to Question 15.06.03-4:

The steam generator (SG) apex in the U.S. EPR FSAR, Tier 2, Figure 15.6-26—SGTR Event - Integrated Mass Flashed refers to the boiler region volume that contains the top (apex) of the U-tubes.

FSAR Impact:

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

U.S. EPR Final Safety Analysis Report Markups

15.0.0.3.8 Limiting Single Failures

The accident analyses presented in Chapter 15 incorporate the most limiting active single failure of a safety-related system. Table 15.0-11—Single Failures Assumed in the Accident Analysis lists the most limiting single failure for each event. Passive failures are not considered, except as event initiators, during the first 24 hours of the event. The following pieces of equipment are considered either as passive devices or are designed to be single failure proof and, therefore, are not subject to single failure:

- Main steam safety valves (MSSVs).
- Pressurizer safety relief valves (PSRVs), when actuated by a spring-driven pilot. A single failure is considered when the PSRVs are switched to the electrically driven solenoids that reduce their opening setpoints for low-temperature overpressure protection (LTOP).
- Main steam relief isolation valve (MSRIV), normally closed. This valve is designed to be single-failure proof. Maintenance on the actuating solenoids is limited by TSs.

A loss-of-offsite power (LOOP) and a stuck RCCA are not considered single failures. A stuck RCCA is incorporated into the RT reactivity insertion. LOOP is incorporated whenever it makes the event more severe.

Operator errors are considered as potential single failures. An operator error is considered as a potential single failure for actions expected or directed by emergency procedure, e. g., failure to redirect EFW following FWLB. Operator error is not considered a potential single failure for actions that are not expected or directed by procedure, e.g., safety injection system (SIS) termination following a legitimate safety injection (SI) signal.

15-8

15.0.0.3.9 Overview of the Incore Transient Methodology

The Low DNBR Channel and High LPD Channel Limiting Safety System Setting (LSSS) are designed to monitor the local behavior of departure from nucleate boiling (DNB) and linear power density (LPD) using incore self-powered neutron detectors (SPNDs), rather than inferring it from excore power measurement. The term "incore trips" is used to represent these two trips. Additionally, there are DNB and LPD Limiting Condition for Operation (LCO) functions used for monitoring purposes, which also utilize the incore SPND signals.

DNBR Protection

The minimum departure from nucleate boiling ratio (DNBR) at any point in the core during anticipated operational occurrence (AOO) events must be restricted to maintain the integrity of the fuel rod barriers to radionuclide release. This protection is afforded by the Low DNBR Channel LSSS and the DNB LCO, in conjunction with other Protection System (PS) trips and LCO functions.

Low DNBR Channel

The Low DNBR Channel LSSS setpoints are established such that the point of minimum DNBR in the core will not experience DNB, at 95 percent probability and with 95 percent confidence. The DNBR trip limits are based upon (1) the point at which DNB occurs, and (2) uncertainties affecting the trip. The latter encompasses uncertainties related to:

- Process variable measurement (temperature, flow, pressure, and power).
- Critical heat flux correlation.
- Online DNBR algorithm.
- Assembly and rod bow.

The DNBR trip is based upon the evaluation of a closed-channel model in the plant computer. This model is adjusted in design calculations to provide DNBR predictions in close agreement with those from the approved sub-channel analysis code, LYNXT. Deviations in these DNBR predictions are accommodated as allowances in the setpoint established for the trip.

If the Low DNBR Channel LSSS cannot resolve the degradation in DNBR during a transient event, the combination of the DNB LCO and other PS trips are used to provide protection against DNB. The Low DNBR Channel LSSS is activated at all power levels above the P2 permissive setting (approximately 10 percent power), and is credited in all safety analysis calculations initiated above that power level. The AOO events that provide the basis for the Low DNBR Channel trip are:

- Decrease in Feedwater Temperature.
- Increase in Feedwater Flow.
- Increase in Steam Flow.
- Inadvertent Opening of a Steam Generator Relief or Safety Valve.
- Uncontrolled Control Rod Assembly Withdrawal at Power.
- Control Rod Misoperation (System malfunction or operator error).
- Inadvertent Decrease in Boron Concentration in the Reactor Coolant System.
- Inadvertent Opening of a Pressurizer Relief or Safety Valve.

Although not specifically designed to intercede in postulated accidents (PA), the Low DNBR Channel LSSS may mitigate the radiological consequences of DNB-challenging PA events in which the DNB degradation can be resolved; for example, Main Steam Line Break (Section 15.1.5).

The Low DNBR Channel setpoints are established in statistical setpoint calculations using the methodology in Reference 2, considering static conditions. Safety analysis calculations consider dynamically compensated conditions, and are designed to demonstrate the adequacy of the trip compensation settings. If the combination of the trip compensation settings and the statically established setpoints are not sufficient to protect the specified acceptable fuel design limit (SAFDL), then either the compensation settings and/or the trip setpoints are adjusted to afford that protection. Because the DNB LCO is credited as an initial condition at the initiation of trip-basis events, the DNB LCO settings may alternatively be adjusted to provide additional initial DNB margin.

At power levels below the P2 permissive, the Low DNBR Channel LSSS is not active. Therefore, for safety analysis events initiated below this power level, a deterministic evaluation of the DNB performance during the event is performed directly with the approved sub-channel analysis code LYNXT as described in Section 4.4.4.5.2.

In safety analysis evaluations in which the Low DNBR Channel is active and predicted to afford primary protection, the compensation settings on the trip are examined to confirm that the SAFDL on DNB is not violated. For cases protected by other trips, the transient Δ(DNBR) allowance is examined to confirm it does not exceed that considered in the DNB LCO setpoint.

DNB LCO Setpoint

The DNB LCO function, in conjunction with PS trips and other LCO functions, protects against events in which the Low DNBR Channel LSSS cannot resolve DNB margin degradation. This protection is afforded by imposing a minimum allowable DNBR threshold during steady-state operation, below which the plant cannot operate. The amount of initial DNBR margin represented by these limits is sufficient to accommodate the transient degradation in DNBR prior to the intercession of a PS trip. The DNB LCO is credited in safety analysis as a restriction on the initial conditions permissible at the initiation of a transient event. The uncertainties considered in the DNB LCO setpoint are similar to those of the Low DNBR Channel LSSS.

Potentially limiting events that are protected in part by the DNB LCO are:

- Increase in Steam Flow.
- Loss of Forced Reactor Coolant Flow (Partial Loss).

- Loss of Forced Reactor Coolant Flow (Full Loss).
- Uncontrolled Control Rod Assembly Withdrawal at Power.

A Δ(DNBR) of 0.60, which bounds the Complete Loss of Forced Reactor Coolant Flow (Section 15.3.2) event, forms the basis for the DNB LCO settings credited in the safety analysis.

LPD Protection

The maximum LPD at any point in the core during AOO events must be restricted to maintain the integrity of the fuel rod barriers to radionuclide release. This protection is afforded by the High LPD Channel LSSS and the LPD LCO, in conjunction with other PS trips and LCO settings.

High LPD Channel LSSS

The High LPD Channel LSSS setpoints are established such that the point of maximum LPD in the core will not experience either fuel centerline melt (FCM) or excessive cladding strain during trip-basis AOO events, at 95 percent probability and with 95 percent confidence. The trip LPD limit is based upon (1) an LPD value that conservatively represents the threshold at which FCM or clad strain limits are violated, and (2) uncertainties affecting the trip. The former is obtained from the approved fuel rod response code COPERNIC described in Table 4.1-2, which correlates local power density limits to fuel centerline temperature and clad strain limits. The latter encompasses uncertainties related to:

- Local power measurement.
- Variability in LPD due to fuel pellet manufacturing tolerances.
- Assembly and rod bow.

If the High LPD Channel LSSS cannot resolve the degradation in LPD during a transient event, the combination of the LPD LCO and other PS trips are used to afford protection against FCM and clad strain. The High LPD Channel LSSS is activated at all power levels above the P2 permissive setting (approximately 10 percent power), and is credited in safety analysis calculations initiated above that power level. Trip-basis AOO events for the High LPD Channel trip are:

- Increase in Steam Flow.
- Uncontrolled Control Rod Assembly Withdrawal at Power.
- Control Rod Misoperation (System malfunction or operator error).
- Inadvertent Decrease in Boron Concentration in the Reactor Coolant System.

- Inadvertent Opening of a Pressurizer Relief or Safety Valve.

Although not specifically designed to intercede in PA events, the High LPD Channel LSSS may mitigate the radiological consequences of overpower PA events such as the Main Steam Line Break (Section 15.1.5) or Control Rod Ejection (Section 15.4.8).

The setpoints are established in statistical setpoint calculations using the methodology in Reference 2, considering static conditions. Safety analysis calculations consider dynamically compensated conditions, and are designed to demonstrate the adequacy of the trip compensation settings. If the combination of the trip compensation settings and the statically established setpoints are not sufficient to protect the SAFDL, then either the compensation settings and/or the trip setpoints are adjusted to afford that protection. Since the LPD LCO is credited as an initial condition at the initiation of trip-basis events, the LPD LCO settings may alternatively be adjusted to provide additional initial LPD margin.

In safety analyses in which the High LPD Channel is active and affords primary protection, the compensation settings on the trip are evaluated to protect the SAFDL. At power levels below the P2 permissive, the High LPD Channel LSSS is not active. Therefore, for safety analysis events initiated below this power level, deterministic calculations of the maximum LPD are examined to confirm that the SAFDL is not violated. For cases protected by other trips, the transient Δ(LPD) allowance is evaluated in relation to the LPD LCO setpoint.

LPD LCO

The LPD LCO function, in conjunction with PS trips and other LCO functions, protects against events in which the High LPD Channel LSSS cannot resolve LPD margin degradation. This protection is afforded by imposing a maximum allowable local LPD threshold during steady-state operation, above which the plant cannot operate. The amount of initial LPD margin represented by these limits is sufficient to accommodate the transient degradation in LPD prior to the intercession of a PS trip. The LPD LCO is credited in safety analysis as a restriction on the initial conditions permissible at the initiation of a transient event.

The LPD LCO setpoints are determined by combining uncertainties about the minimum of (1) the steady-state LPD credited in Loss of Coolant Accident (LOCA) calculations, and (2) the transient LPD limit less the maximum transient LPD degradation for any LCO-basis event. The uncertainties considered in the LPD LCO setpoint are similar to those of the High LPD Channel LSSS.

Potentially limiting events that are protected in part by the LPD LCO are:

- Increase in Steam Flow.

- Steam System Piping Failures Inside and Outside of Containment.
- Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition.
- Uncontrolled Control Rod Assembly Withdrawal at Power.
- Spectrum of Rod Ejection Accidents.
- Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.

Transient Analysis with Incore Trips

The transient analysis is performed with incore trip models decoupled from the system simulation code, S-RELAP5. The incore trip models are generically referred to as the "algorithm" or separately as the Low DNB Channel algorithm and High LPD Channel algorithm. The core boundary conditions for the algorithm are generated in S-RELAP5 and power distributions are generated in the nodal neutronics code, PRISM.

The Low DNB Channel and High LPD Channel algorithms are simulated to predict times at which the incore trip setpoints are reached, and to demonstrate the adequacy of the dynamic compensation on the trips. Table 15.0-7 lists the incore trip setpoints used in the accident analyses. The methodology for confirming the dynamic compensation is described in Section 9.4 of Reference 2.

The Low DNB Channel and High LPD Channel algorithm use the following measurements:

- The reactor power distributions derived from the SPNDs, which are part of the nuclear incore instrumentation.
- The primary system pressure derived from the primary pressure sensors.
- The core flow derived from the reactor coolant pump (RCP) speed sensors and the calibrated volumetric flow from a surveillance measurement.
- The reactor inlet temperature derived from the cold-leg temperature sensors.

15.0.0.3.10 Plant Design Changes

The information presented in Section 15.0 represents the current U.S. EPR design. Some of the analyses presented in this section used slightly different values. In these cases the differences have been evaluated and found to have a negligible or conservative impact on the results and conclusions.

15.01.01 - 15.01.04-1

The low departure from nucleate boiling (DNB) channel algorithm and high linear power density (LPD) channel algorithm is used to predict the RT and confirm the adequacy of the dynamic compensation of the algorithm consistent with the Incore Trip Setpoint and Transient Methodology for the U.S. EPR (Reference 2).

The following assumptions apply to the analysis of the decrease in feedwater temperature event:

- At hot full power (HFP), a feedwater temperature reduction of 100°F is assumed to occur instantaneously at the feedwater inlet to each SG. This condition conservatively bounds possible physical reduction in temperature attributable to a single malfunction.
- A bounding end-of-cycle (EOC) MTC value is used.
- No operator actions are credited.

The applicable acceptance criteria for the decrease in feedwater temperature event are as follows:

- The thermal margin limit SAFDLs, DNB and fuel centerline melt (FCM), criteria are satisfied.
- Maximum RCS pressure limits are not exceeded.

15.01.01 - 15.01.04-1

This event meets the SAFDL criteria, which are satisfied by the combination of the low DNB and high LPD limiting conditions for operation (LCO) and RT setpoint described in Reference 2.

15.1.1.3 Results

Table 15.1-1—Decrease in Feedwater Temperature - Key Input Parameters provides the initial conditions for this event. Table 15.1-2—Decrease in Feedwater Temperature - Key Equipment Status lists parameters for the protection system functions that may mitigate this event. Table 15.1-3—Decrease in Feedwater Temperature - Sequence of Events provides the sequence of events. Figure 15.1-1—Decrease in Feedwater Temperature - Main Feedwater Flow Rate through Figure 15.1-8—Decrease in Feedwater Temperature - Pressurizer Level show the plant response for a representative decrease in feedwater temperature case. Figure 15.1-57—Decrease in Feedwater Temperature - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

15.01.01 - 15.01.04-1

The transient is initiated by instantly decreasing the main feedwater (MFW) temperature to the SGs to 346°F from the normal HFP temperature of 446°F. The decrease in feedwater temperature in the SGs causes a decrease in cold-leg temperature which, in combination with a negative MTC, increases core power. This condition quickly generates a low DNB ratio (DNBR) RT and subsequent turbine trip (TT). The TT closes the turbine stop valves (TSV~~s~~) and is assumed to cause a loss of offsite power (LOOP). The LOOP terminates MFW flow and starts the coastdown of the reactor coolant pumps (RCP~~s~~). The closing of the TSVs causes SG secondary pressure to increase to the main steam relief train (MSRT) setpoint, which reduces heat removal from the RCS. This condition causes RCS temperatures and pressure to increase. As the RCPs coast down, primary system temperatures increase further to establish natural circulation, which contributes to the increase in RCS pressure. The pressurizer (PZR) safety relief valves (PSRV~~s~~) open to control RCS pressure. The plant is now in a stable, controlled state.

15.1.1.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs, nor is there a release of radioactive materials to the environment.

15.01.01 - 15.01.04-1

15.1.1.5 Conclusions

~~The DNB RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm, are adequate to protect the DNB SAFDL for the conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL. The maximum linear power density (LPD) realized is below the limit that precludes FCM. Therefore, fuel clad integrity is maintained.~~ The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient.

15.1.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.1 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

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

to the turbine. Secondary pressure decreases and thereby increases overall heat transfer to the SGs.

A quasi-steady-state is established until the high SG level RT signal is reached. This condition terminates the event automatically by closing the MFW isolation valve. Should that isolation valve fail, the HL and LL isolation valves also close on this signal.

The increase in feedwater flow event is classified as an AOO (described in Section 15.0.0.1) that is expected to occur with moderate frequency. The event potentially challenges SAFDLs. It also can cause overfill of the SG, which could force liquid water into the steam lines and potentially create a more serious plant condition.

15.1.2.2 Methods of Analysis and Assumptions

The analysis of the increase in feedwater flow event uses the approved non-LOCA analytical methodology described in Reference 1. The S-RELAP5 computer code (described in Section 15.0.2.4) is used to simulate the event and contains provisions to model the primary and secondary systems, as well as the core reactivity response.

15.01.01 - 15.01.04-1

The low DNB channel algorithm and high LPD channel algorithm are~~is~~ simulated to predict RT and adequacy of the dynamic compensation of the algorithm consistent with Reference 2. The focus for this event is meeting SAFDLs. The ~~DNB~~ SAFDLs are ~~is~~ satisfied by the combination of the low DNB and high LPD LCO and RT setpoint described in Reference 2.

15.01.01 - 15.01.04-1

The analysis of the event assumes turbine steam demand remains constant, i.e., the turbine valves move to keep the steam flow to the turbine constant. A spectrum of scenarios is analyzed at HFP, 60 percent and 25 percent of rated thermal power, and hot zero power (HZP) at BOC and EOC conditions. Cases at BOC are analyzed with action by the non-safety-related ACT control function because this condition makes the results for BOC cases more severe. Cases at EOC conditions are analyzed with and without action by the non-safety-related ACT control function because this condition makes the results more severe.

For the purposes of analyzing this event, the most severe single failure is the failure of the HL isolation valve to close on the affected SG. There is no impact on the results of the transient analysis because there is an MFW isolation valve upstream of the HL isolation valve that also closes to terminate the MFW flow.

The following conservative assumptions apply to this increase in feedwater flow analysis:

- The temperature of the MFW to the four SGs is reduced 100°F to account for one string of high-pressure feedwater heaters being out of service.

15.01.01 - 15.01.04-8

- MFW flow to the affected SG increases instantaneously to a bounding 150 percent of the HFP flow rate, regardless of initial power level.
- SG tube plugging is neglected in order to maximize heat transfer from the RCS to the SGs.
- A bounding EOC negative MTC is used to maximize the increase in power as moderator temperature decreases.
- The SG high-level setpoint for RT and isolation of MFW is biased to 100 percent of the narrow range sensor, which bounds the maximum measurement uncertainty.

15.1.2.3 Results

15.01.01 - 15.01.04-1

This event does not challenge SAFDLs or overpressure criteria. Presented as a representative case, the scenario that causes the highest core power is a BOC HFP scenario in which the non-safety-related ACT control function is simulated. Table 15.1-4—Increase in Feedwater Flow - Key Input Parameters presents the initial conditions for this event. Table 15.1-5—Increase in Feedwater Flow - Key Equipment Status presents the status of equipment for this event. Table 15.1-6—Increase in Feedwater Flow - Sequence of Events provides the sequence of events. Figure 15.1-9—Increase in Feedwater Flow - Reactor Power through Figure 15.1-21—Increase in Feedwater Flow - Reactivity show the plant response. Figure 15.1-58—Increase in Main Feedwater Flow - Representative Plot of Normalized DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The transient is initiated by instantaneously increasing MFW flow to the affected SG to 150 percent of HFP flow rate. This immediately causes a rising water level in the affected SG and an increase in the heat transfer rate, but a reduction in steam production in that SG. Steam flow from the unaffected SGs increases to compensate and maintain a constant steam flow to the turbine as the TCV opens. The increase in heat transfer, particularly in the economizer region, causes the affected loop cold-leg temperature to decrease. Because the scenario is at BOC, the colder water reaching the core does not cause core power to increase.

After about 25 seconds, the ACT control function begins to withdraw control rods to compensate for the reduction in RCS temperatures. Power rises slowly as the ACT control function adds reactivity. The increase in power causes negative reactivity insertion because of Doppler feedback. These two effects tend to counter each other.

The high SG level trip signal occurs at 187 seconds, thereby isolating MFW and terminating the transient. The resulting TT is assumed to cause a LOOP and start the coastdown of the RCPs. RCS temperature and pressure increase as secondary system pressure increases to the MSRT setpoint and the RCS system transitions to natural circulation. The PSRVs open to control RCS pressure. The plant is now in a stable controlled state.

15.1.2.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs, nor is there a release of radioactive materials to the environment.

15.1.2.5 Conclusions

15.01.01 - 15.01.04-1

~~The DNB RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm, are adequate to protect the DNB SAFDL for the conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL. The maximum LPD realized is below the limit that precludes FCM. Therefore, fuel clad integrity is maintained.~~The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degredation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient.

15.1.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.2 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.2.5.
2. Fuel cladding integrity shall be maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.
   - Response: Fuel cladding integrity is maintained as concluded in Section 15.1.2.5.

D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

– Response: The setpoints for the mitigating systems include instrument uncertainty.

15.1.3 Increase in Steam Flow

15.1.3.1 Identification of Causes and Event Description

The increase in steam flow event is an AOO that occurs when main steam flow is increased above the steady-state demand. The magnitude can range from a small increase in flow caused by the opening of the turbine control valves to a large increase caused by the simultaneous opening of the turbine bypass valves. The increase in flow can be initiated by operator action or failure of a control system that results in spurious operation of the bypass system or turbine control valves.

15.01.01 - 15.01.04-1

A small increase in main steam flow can cause the RCS to stabilize at a higher core power configuration that does not cause an RT. A greater increase in steam flow produces a power excursion that is sufficient to produce a low DNBR trip or high LPD trip that terminates the event. For even larger increases in steam flow, the rapid drop in pressure in the main steam lines trips the reactor on the high SG pressure drop trip within a few seconds. The rapid trip precludes a large power excursion.

15.1.3.2 Methods of Analysis and Assumptions

The analysis of the increase in steam flow event uses the approved non-LOCA analytical methodology described in Reference 1. The S-RELAP5 computer code (described in Section 15.0.2.4) is used to simulate the event and contains provisions to model the primary and secondary systems, as well as the core reactivity response. The low DNB channel algorithm and high LPD channel algorithm are ~~is~~ used for predicting RT and the adequacy of the dynamic compensation of the algorithm consistent with Reference 2.

15.01.01 - 15.01.04-1

A spectrum of steam flow increase scenarios is analyzed at HFP, 25 percent of rated thermal power and HZP at BOC and EOC conditions. The maximum flow increase analyzed corresponds to the opening of the six turbine bypass valves. A spectrum of flow rates is analyzed corresponding to the opening of one through six of the bypass valves. The conditions for the representative case described in Section 15.1.3.3 are described below.

The following conservative assumptions apply to the increase in steam flow analysis:

- Respective bounding values of MTC are used for BOC and EOC.

- Cases are analyzed with and without action by the non-safety-related ACT control function because this condition could make the results more severe at BOC.
- A 100°F reduction in MFW temperature is assumed, consistent with a high-pressure MFW heater train out of service.
- The only mitigating equipment credited is the PS, which is designed as single failure proof. There is no single failure that makes the consequences of this event more severe.
- No operator actions are assumed.

15.1.3.3 Results

15.01.01 - 15.01.04-1

Table 15.1-7—Increase in Steam Flow - Key Input Parameters provides the initial conditions for this event. Table 15.1-8—Increase in Steam Flow - Key Equipment Status lists parameters for the protection system functions that may mitigate this event. Table 15.1-9—Increase in Steam Flow - Sequence of Events provides the sequence of events. Figure 15.1-22—Increase in Steam Flow - Reactor Power through Figure 15.1-32—Increase in Steam Flow - Reactivity (Detail) show system response to a representative increase in steam flow case. Figure 15.1-59—Increase in Steam Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The event is initiated by simulating the simultaneous opening of the six turbine bypass valves. This causes a rapid decrease in pressure in the SGs. The decreased pressure increases heat transfer from the RCS to the SG. This causes the RCS cold-leg temperatures to decrease and reactor power to increase because of the highly negative MTC. The decrease in RCS coolant temperature causes the ACT control function to withdraw control rods to attempt to raise the RCS average temperature. Rods are withdrawn for about two seconds and reactor power continues to increase until the event is terminated by a low DNBR RT.

The resulting TT is assumed to cause a LOOP that terminates MFW flow and starts the coastdown of the RCPs. The increase in secondary system pressure to the MSRT setpoint and the transition to natural circulation causes the heatup and pressurization of the RCS. The PSRVs open to control RCS pressure. The plant is now in a stable controlled state.

15.1.3.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs, nor is there a release of radioactive materials to the environment.

15.01.01 - 15.01.04-1

15.1.3.5 Conclusions

~~The DNB SAFDL is satisfied by the combination of the low DNB LCO and RT setpoint described in Reference 2. The dynamic compensation of the low DNB channel algorithm is shown to be adequate to protect the SAFDL when the RT setpoint is reached. No DNB or FCM concerns apply to HZP cases. High linear power density (HLPD) limits are not exceeded, which demonstrates that FCM is prevented.~~ The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to isse an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient.

15.1.3.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.3 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

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

   – Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.3.5.

2. Fuel cladding integrity is maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.

   – Response: Fuel cladding integrity is maintained as concluded in Section 15.1.3.5.

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

   – Response: The event is mitigated and the plant is maintained in a stable condition through the automated response of safety-related equipment, thus avoiding an aggravated plant condition.

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

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Event Description

The inadvertent opening of the SG relief or safety valve is an AOO that increases steam flow and thus causes a mismatch between the energy being generated in the reactor core and the energy being removed by the secondary system. This condition causes a cooldown of the primary system. An increase in power occurs if the moderator temperature reactivity feedback coefficient is negative or if the rod control system is in an automatic mode and begins to remove control rods from the reactor core in response to the decrease in RCS temperature resulting from the increased steam flow. If the power increase is sufficiently large, a low DNBR RT or high LPD RT is initiated.

15.01.01 - 15.01.04-1

Overpressure protection is provided on each of the four main steam lines by an MSRT and two main steam safety valves (MSSVs). Each MSRT consists of a single-failure proof main steam relief isolation valve (MSRIV) and a downstream main steam relief control valve (MSRCV). The discharge capacity of each MSRT is approximately 50 percent of the full load steam flow of the SG. The discharge capacity of each MSSV is approximately 25 percent. Although the inadvertent opening of a relief valve occurs in one SG, the four SGs pick up and share the extra load within a few seconds because their steam lines are connected at the turbine bypass header. Consequently, conditions in the RCS remain symmetric until RT and closure of the MSIVs.

The MSRIVs are fast opening valves that are normally closed. The MSRCVs are normally open control valves. They are fully open at HFP and close to 40 percent as core thermal power decreases to 40 percent, below which they remain 40 percent open. The opening of the MSRIV automatically switches the MSRCV to control SG pressure to a high relief valve setpoint. It closes when SG pressure falls below that setpoint.

When the low SG pressure or high SG pressure decrease setpoint is reached, the PS initiates RT and closes the MSIVs. This isolates the affected SG from the other three unaffected SGs. When the low-low SG pressure or high-high SG pressure drop setpoint is reached, the PS isolates the MSRIV and the low-load feedwater line in the affected SG.

Should an MSRIV inadvertently open as postulated, the associated MSRCV terminates steam flow through the affected MSRT automatically when SG pressure decreases below its setpoint. The MSRCV requires 40 seconds to close. The automatic closure of the MSRCV in the affected MSRT terminates the event, but is postulated to fail open as the most severe single failure. This allows the blowdown of the affected SG to continue until dryout or closure of the MSRIV on the low-low SG pressure PS signal. The MSRIV might cycle between its opening and closing setpoints until the operator initiates a cool down in the unaffected SGs to transition to RHR cooling. Once RCS temperature decreases, the MSRIV in the affected SG remains closed.

The evolution of the stuck open MSSV scenario is similar to that of the inadvertent opening of MSRIV except for two differences: the capacity of the MSSV is half that of the MSRT and the MSSV cannot be isolated.

The inadvertent opening of an MSRIV and an MSSV are both AOOs evaluated for the following acceptance criteria: 15.01.01 - 15.01.04-1

- The event challenges SAFDLs by DNBR and LPD prior to RT and, in the long-term by a return to criticality because of the continued cooldown of the RCS.
- The event also challenges radiological release limits.

15.1.4.2 Methods of Analysis and Assumptions

The analysis of the inadvertent opening of the SG relief valve or safety valve events uses the approved non-LOCA analytical methodology described in Reference 1. The S-RELAP5 computer code, described in Section 15.0.2.4, is used to simulate the event. This code simulates the primary and secondary systems, as well as the core reactivity response. The low DNB channel algorithm and high LPD channel algorithm is used for predicting the RT and the adequacy of the dynamic compensation of the algorithm consistent with Reference 2. The inadvertent opening of the MSSV and MSRIV are different enough scenarios that both are evaluated. Although the MSSV has half the capacity of an MSRT, it cannot be isolated. 15.01.01 - 15.01.04-1

The following conservative assumptions apply to the analysis of an inadvertent opening of an MSRIV:

- The initiating failure is a spurious signal that opens the MSRIV. This failure does not prevent its later closing when it receives a signal from the PS to close.
- The most severe single failure is the failure of the associated MSRCV to close and terminate the event.
- The temperature of the MFW to the four SGs is reduced to account for one string of high-pressure feedwater heaters being out of service.
- Respective bounding values of MTC are used for BOC and EOC.
- Cases are analyzed with and without action by the non-safety-related ACT control function because it could make the results more severe at BOC.

The analysis of the inadvertent opening of an MSSV is the same as for the inadvertent opening of an MSRIV except:

- The MSSV is not isolatable.

- The most severe single failure is the failure of an MSRCV to close in another SG.

15.1.4.3 Results

15.1.4.3.1 Inadvertent Opening of an MSRIV

15.01.01 - 15.01.04-1

This event is mitigated by the low DNBR RT or the high LPD RT. The scenario that generates the earliest RT is presented as a representative case. This is an EOC HFP scenario in which the non-safety-related ACT control function is deactivated. Table 15.1-10—Inadvertent Opening of an SG Relief or Safety Valve - Key Input Parameters provides the initial conditions for this case. Table 15.1-11—Inadvertent Opening of an SG Relief or Safety Valve - Key Equipment Status presents the status of equipment. Table 15.1-12—Inadvertent Opening of an SG Relief or Safety Valve - Sequence of Events presents the sequence of events. Figure 15.1-33—Inadvertent Opening of a SG Relief or Safety Valve - MSRT Flow Rate through Figure 15.1-43—Inadvertent Opening of a SG Relief or Safety Valve - TSV Position show system response for a representative case for an inadvertent actuation of the MSRT. Figure 15.1-60—Inadvertent Opening of a SG Relief or Safety Valve - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

15.01.01 - 15.01.04-1

The transient is initiated by instantaneously fully opening the MSRIV in the affected SG. The MSRCV already is fully open during HFP operation. This condition causes a rapid decrease in pressure in the affected SG as well as the unaffected SGs via the common steam line header, which increases heat transfer from the RCS. As cold-leg temperatures decrease, reactor power increases because of the highly negative MTC. The reactor power continues to increase until the event is terminated by RT on a low DNBR signal. The subsequent TT is assumed to cause a LOOP that terminates MFW flow and starts the coastdown of the RCPs. Following TT, pressure in the secondary system initially increases as the TSVs close. Continued steam flow from the open MSRT train cools both the secondary system and, thereby, the RCS. This cooling causes SG and RCS pressures to start decreasing about 10 seconds after RT.

At about 100 seconds, an SG high-pressure drop signal is generated and the PS closes the MSIVs. This action isolates the three unaffected SGs from the affected SG. The affected SG continues to blow down until the SG low-low pressure setpoint is reached that closes the MSRIV. The MSRIV in the affected SG continues to slowly cycle open and closed between PS setpoints until the operator initiates a cool down in the unaffected SGs to transition to RHR cooling. Once RCS temperature decreases, the MSRIV in the affected SG remains closed. The unaffected SGs provide long-term heat removal for core decay heat. The plant enters a stable controlled state. The unaffected SGs provide long-term heat removal for core decay heat. The plant enters a stable controlled state.

~~The DNB RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm, are adequate to protect the DNB SAFDL for the conditions that cause the low DNB channel to issue an RT. The high HLPD limits are not exceeded, which demonstrates that FCM is prevented.~~ The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to isse an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

This scenario also is analyzed at EOC HZP to determine the post-RT peak return to power. The analysis shows that the peak return to power for this event is 314 MW—a power level that does not cause fuel damage. Peak steam system and RCS pressures do not exceed 110 percent of design value.

15.1.4.3.2 Inadvertent Opening of an MSSV

The inadvertent opening of an MSSV scenario that has the greatest post-RT return to power is an EOC HZP case. The analysis shows a post-RT return to a power to 216 MW. This does not cause fuel damage. This scenario provides thermal-hydraulic conditions for evaluating the radiological consequences of long-term steam release through the unisolable failed-open MSSV.

The inadvertent opening of an MSSV event is bounded by the inadvertent opening of an MSRT event in regard to SAFDLs.

15.1.4.4 Radiological Consequences

The inadvertent opening of an MSSV is the limiting AOO event for radiological consequences. The dose acceptance criteria for such events are defined in 10 CFR Part 50, Appendix I, for the summation of radioactive releases during normal operation and the annual average radioactive releases due to an AOO event based on realistic assumptions. The RCS and secondary coolant concentrations correspond to normal operating conditions, and are determined through application of the ANSI/ANS-18.1-1999 standard (Reference 4).

The radiological consequences are determined at a distance of 0.5 mile from the site. The analysis conservatively assumes that the entire ingestion pathway is located at this distance and the exposure is continuous during the entire event. Subsequent exposure continues for an entire year thereafter, accounting for submersion, inhalation, ingestion and ground-shine pathways.

The worst-case organ dose is determined to be to the infant thyroid and to amount to 0.79 mrem, mostly due to the ingestion of milk. This value is based on the dose conversion factors from Federal Guidance Report 11 (Reference 5). These limits are about five percent of the 10 CFR Part 50, Appendix I limit of 15 mrem.

15.01.01 - 15.01.04-1

15.1.4.5 Conclusions

~~The DNB SAFDL is satisfied by the combination of the low DNB LCO and RT setpoint described in Reference 2. The dynamic compensation of the low DNB channel algorithm is shown to be adequate to protect the SAFDL when the RT setpoint is reached. The HLPD limits are not exceeded, which demonstrates that FCM is prevented.~~ The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to isse an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient. Additionally, the analysis of post-RT consequences show that the peak return-to-power value is below the threshold where fuel failure occurs.

15.1.4.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.4 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.4.5.
2. Fuel cladding integrity shall be maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.
   - Response: Fuel cladding integrity is maintained as concluded in Section 15.1.4.5.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

– Response: The setpoints for the mitigating systems include instrument uncertainty.

15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR)

The analysis of postulated failures of steam system piping are described in this section. The most severe scenario from a radiological standpoint is a break located in the valve room outside the Reactor Building but upstream of the MSIVs. Pipe failures inside the Reactor Building are not analyzed because their radiological consequences are less severe.

In accordance with the classification of events described in Section 15.0.0.1, minor steam system piping failures are considered AOOs, and larger breaks are treated as PAs. The analyses described in Section 15.1.5 address large main steam line breaks (MSLB~~s~~), which are PAs.

15.1.5.1 Identification of Causes and Event Description

The rupture of a main steam pipe increases the rate of energy removal from the RCS and lowers RCS temperatures and pressures. Initially, the rate of steam flow in the failed pipe increases, but it subsequently decreases with time as the steam pressure drops. Because the SGs are connected via the turbine bypass header, the four SGs depressurize through the break. The four RCS loops cool down, although the rate is greater in the loop with the affected SG.

15.01.05-8

When the low SG pressure or high SG pressure decrease setpoint is reached, the PS initiates RT and closes the MSIVs. Because the break is assumed to be located upstream of an MSIV, their closure isolates the ~~affected SG~~ unaffected SGs from the break. When the low-low SG pressure or high-high SG pressure decrease setpoint is reached, the PS isolates the low-load feedwater line in the affected SG.

Once the MSIVs close, temperatures in the unaffected loops recover while temperature continues to decrease in the affected loop as the SG continues to blow down. Positive reactivity is inserted because of the assumed negative MTC. This condition erodes core shutdown margin, which can lead to re-criticality.

When the low-low SG level setpoint is reached in the affected SG, the PS actuates EFW to that SG. This action prolongs the release of steam through the break, the cooldown of the RCS, and the associated erosion of the shutdown margin. SAFDLs, including cladding strain, can be challenged if shutdown margin is eroded sufficiently for the reactor to return to power, particularly in conjunction with augmented radial power peaking near the stuck-out rod cluster control assembly (RCCA). Where SAFDLs are exceeded, the fuel is assumed to fail. The resulting fuel failure fractions are used to determine radiological consequences.

The event ends when the operator terminates EFW delivery to the affected SG, after which the RCS gradually reheats and reactor shutdown margin recovers.

15.1.5.2 Methods of Analysis and Assumptions

MSLB is analyzed using the approved computer codes and methods described in Reference 1. The system response to the MSLB event is analyzed using the S-RELAP5 computer code (see Section 15.0.2.4). This computer code simulates neutron kinetics, the RCS, the PZR, the SGs, the main steam lines and valves, and the feedwater system. The S-RELAP5 code computes system variables, including temperatures, pressures, flows, and power level. The LYNXT computer code (refer to Section 15.0.2.3) is used to determine the minimum DNBR for the event, and the PRISM computer code (refer to Section 15.0.2.1) is used to calculate the core power distributions. A spectrum of break size cases is analyzed for conservative combinations of core power levels, break locations (upstream and downstream of MSIV), offsite power conditions and single failures. Small break pre-scram events use the low DNB channel algorithm and high LPD channel algorithm for predicting the RT and the adequacy of the dynamic compensation of the algorithm consistent with Reference 2.

15.01.05-3

The limiting postulated MSLB event scenario is a 1.72 ft^2 break in a main steam line outside the Reactor Building, upstream of the MSIV, at EOC and HZP conditions. The consequences of the event are exacerbated by assuming offsite power remains available to operate the RCPs. Moreover, it is assumed that the MSRCV on one of the unaffected main steam lines fails in the fully open position. With this assumption, when the unaffected SGs repressurize sufficiently following MSIV closure for their MSRIVs to open, the SG with the faulted MSRCV depressurizes more than it might otherwise, until the corresponding MSRIV closure setpoint is reached. This situation exacerbates the RCS cooldown and represents the worst-case single failure.

Additional assumptions for the analysis include the following:

- One train of each mitigating safety system is assumed to be unavailable due to maintenance.
- The highest worth RCCA is assumed to be located in the affected core sector and stuck in the fully withdrawn position, which reduces shutdown margin and augments power peaking.
- EOC conditions are assumed as they yield the most limiting combination of reactivity coefficients and control rod worth.
- Mixing of fluid between core sectors is modeled as described in Reference 1.
- MFW flow is treated conservatively as described in Reference 1.

15.1.5.3 Results

The limiting MSLB event scenario is initiated at EOC HZP conditions by a postulated 1.72 ft^2 break in a main steam line outside the Reactor Building. Offsite power remains available to operate the RCPs. Table 15.1-13—MSLB - Key Input Parameters for Limiting Case presents the input for the limiting case. Table 15.1-14—MSLB - Key Equipment Status for Limiting Case presents the status of key plant equipment and systems. Table 15.1-15—MSLB - Sequence of Events presents the sequence of events for the limiting case. Figure 15.1-44—MSLB - Break Flow Rate through Figure 15.1-56—MSLB - Longer-Term Reactor Power present plots of key system variables. Figure 15.1-61—MSLB (small break, pre-scram) - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

15.01.05-3

The MSLB rapidly reduces pressure in the four SGs generating an SG high pressure drop signal that closes the MSIVs. This closure isolates the three unaffected SGs from the break. As the inventory of the affected SG flashes and is discharged through the break, the RCS cools down, adding positive reactivity. The RCS cooldown is exacerbated once the MSRIVs on the unaffected SGs open, permitting uncontrolled steam release from the unaffected SG, whose MSRCV is assumed to fail in the fully open position. The resultant RCS cooldown and associated reactivity addition causes the reactor to return to critical. As the affected SG approaches dry out, reactor power peaks and begins to decrease. The event is terminated when the operator stops EFW delivery to the affected SG and it dries out. The RCS gradually reheats, reactor shutdown margin recovers and the plant enters a stable controlled condition with heat removal via the unaffected SGs.

The analysis results indicate that the fuel cladding strain limits for AOOs are exceeded for this PA, causing a small number of fuel failures. Table 15.1-16—MSLB - Calculated Fuel Parameters provides the calculated fuel parameters for the limiting case.

15.1.5.4 Radiological Consequences

Radiological consequences for the MSLB event are described in Section 15.0.3.7.

15.1.5.5 Conclusions

The results presented in Section 15.1.5.3 demonstrate that RCS and main steam system pressures are maintained below acceptable design limits. Analysis of the post-RT consequences shows that there is only limited fuel damage. The radiological consequences are within the limits of 10 CFR 50.34(a)(1) and 10 CFR Part 50, Appendix A.

Table 15.1-11—Inadvertent Opening of an SG Relief or Safety Valve - Key Equipment Status

Plant Equipment or System	Status
RT RCCAs	Most reactive RCCA stuck out of core
PZR heaters	Not credited
MSRTs	MSRTs for unaffected SGs available—SG 3~~4~~ MSRCV ~~is~~ fail~~sed~~ ~~open~~ to close
MSIVs	Functional
RCPs	Operating
Feedwater pumps	Operating
EFW	Available
MHSI pumps	One MHSI pump out of service for maintenance; remaining three MHSI pumps available

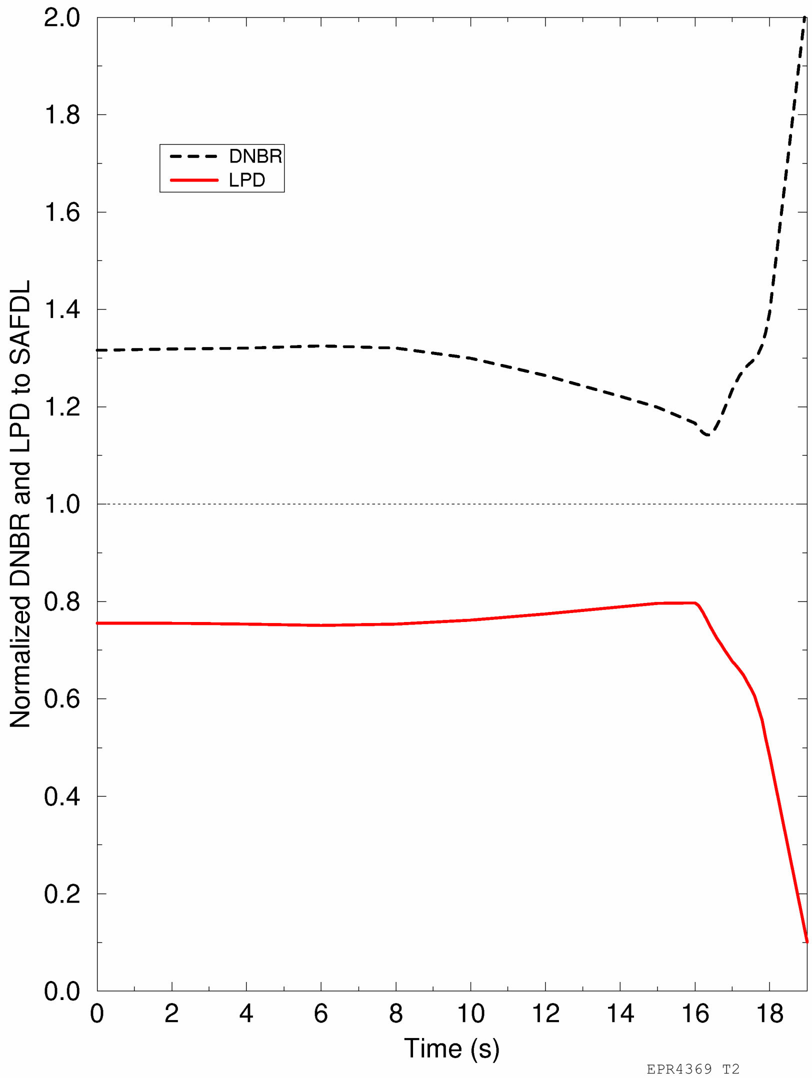
15.01.01 - 15.01.04-11

Table 15.1-16—MSLB - Calculated Fuel Parameters

Parameter	Value
Peak post-scram reactor power	~~649.28~~ 1062.3 MWt
Peak LHGR	~~20.35~~ 19.1 kW/ft
FCM fuel failure	0.00%
Fuel failure at clad strain limit	1.24%

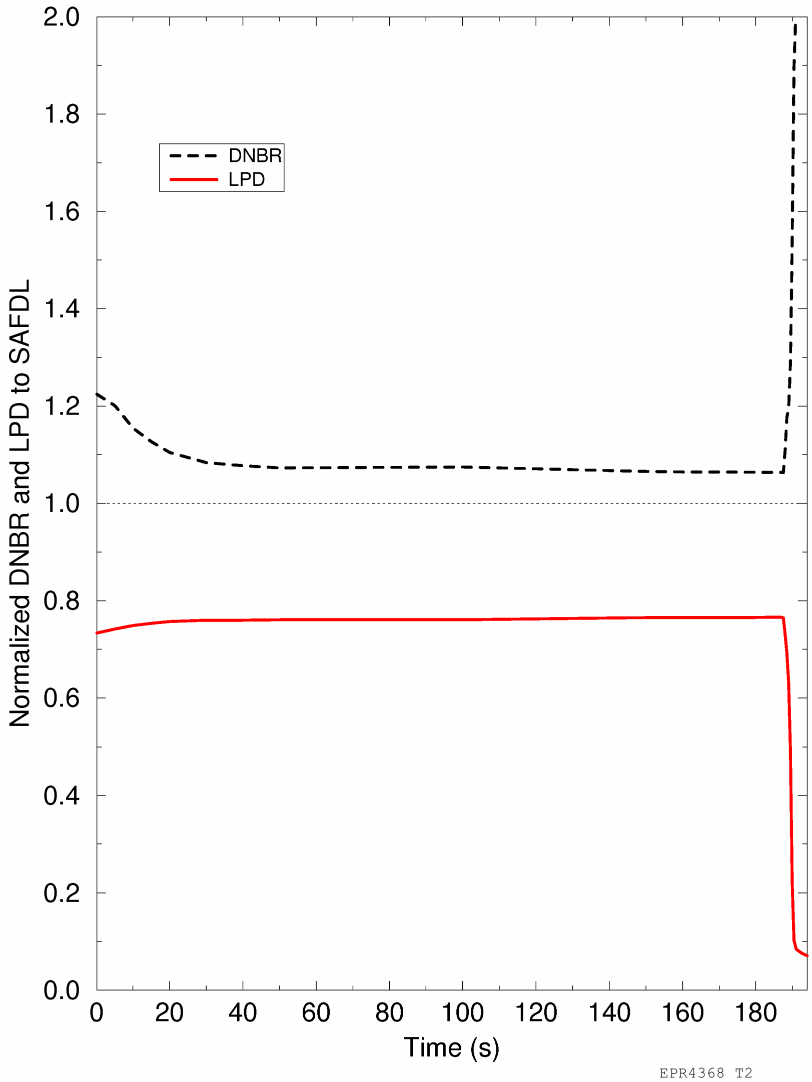
15.01.05-9

Figure 15.1-57—Decrease in Feedwater Temperature - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



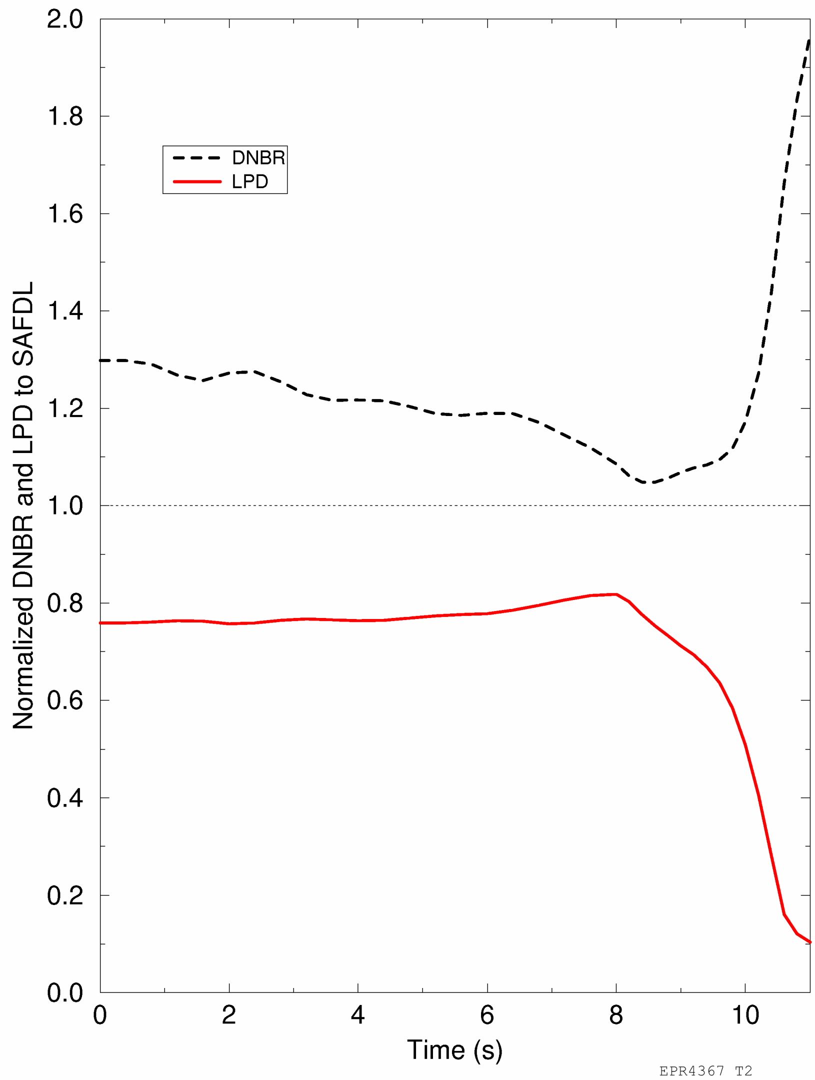
15.01.01 - 15.01.04-1

Figure 15.1-58—Increase in Main Feedwater Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



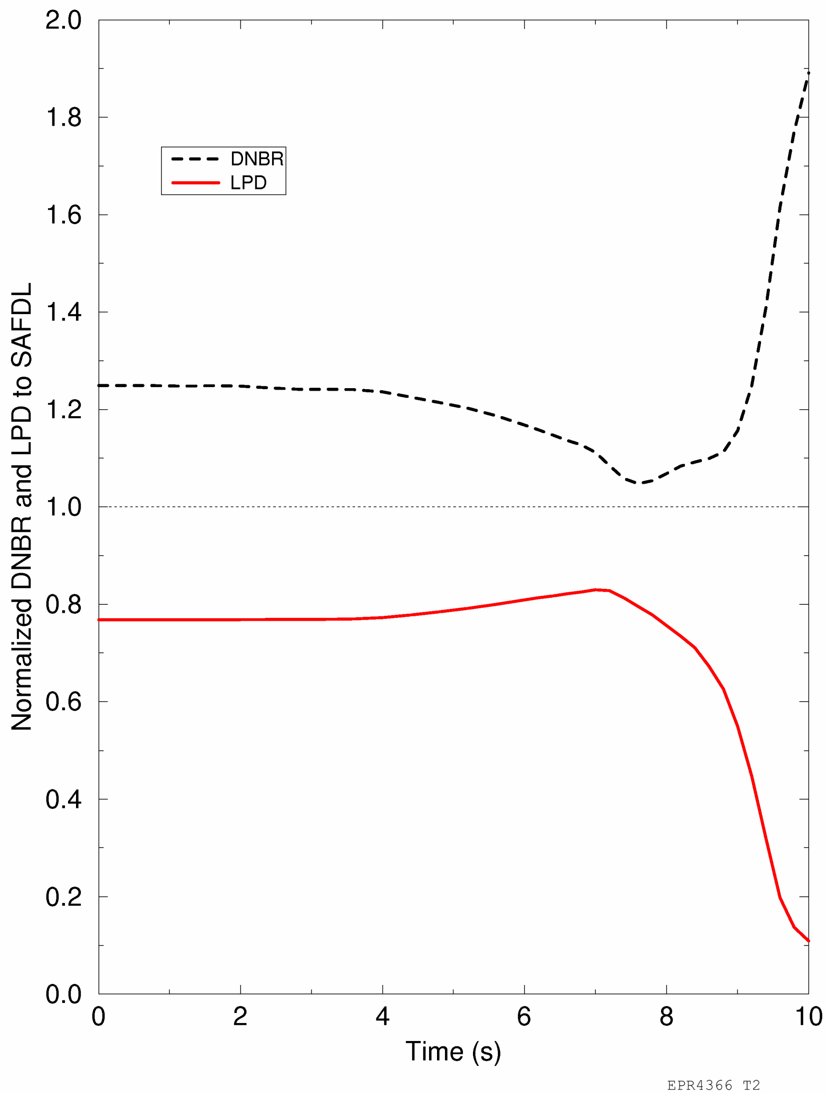
15.01.01 - 15.01.04-1

Figure 15.1-59—Increase in Steam Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



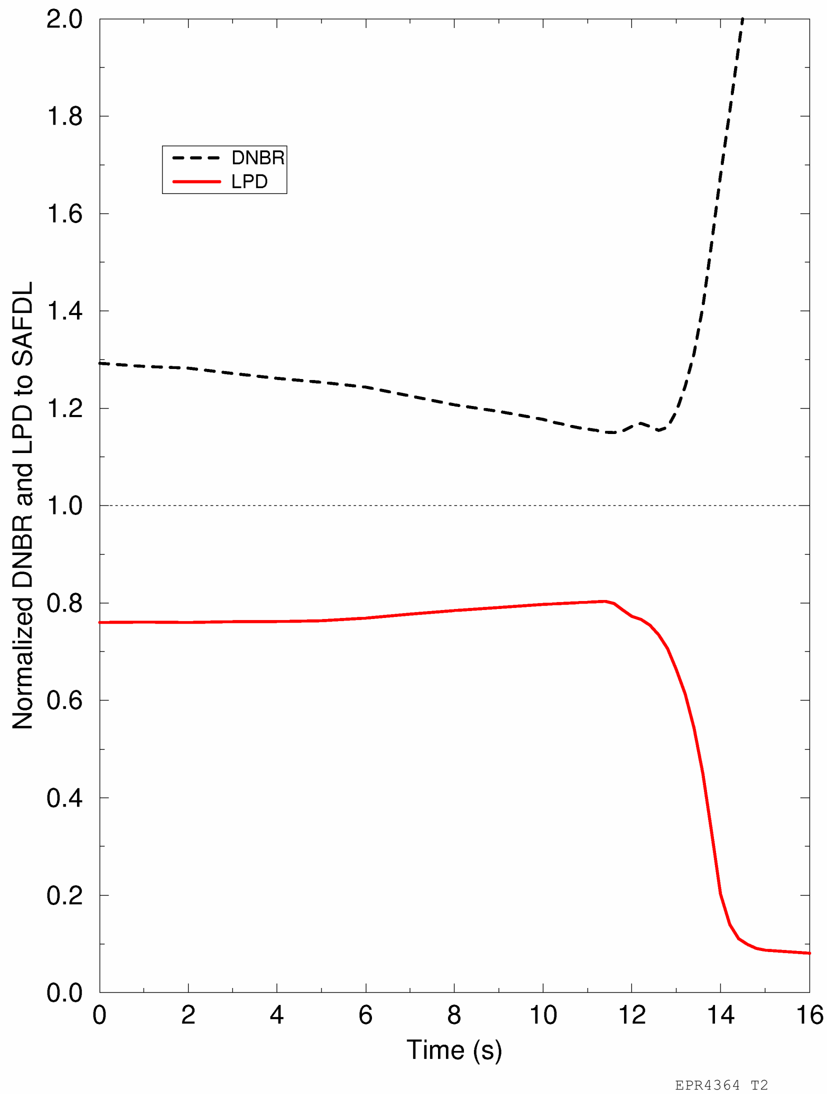
15.01.01 - 15.01.04-1

Figure 15.1-60—Inadvertent Opening of a SG Relief or Safety Valve - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



15.01.01 - 15.01.04-1

Figure 15.1-61—MSLB (small break, pre-scram) - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL



15.01.05-3

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performed using the methodology described in Reference 1. Section 15.0.2 provides a description of the S-RELAP5 analysis method.

15.02.01 - 15.02.05-7

The algorithm, described in Section 15.0.0.3.9, is simulated to predict RT and adequacy of the dynamic compensation of the incore monitoring system in a manner consistent with Reference 3. Additionally, the core thermal-hydraulic computer code LYNXT is used to deterministically calculate the ~~core flow, enthalpy distributions,~~ MDNBR~~, and peak fuel centerline temperatures~~ using the RCS response from S-RELAP5 as a boundary condition. ~~It is described in the Incore Trip Setpoint and Transient Methodology for U.S. EPR (Reference 3). Section 4.4 describes the codes and methods used to evaluate SAFDLs. Section 15.0.3 describes the codes and methods used for the radiological analyses.~~

The MSIVC ~~closure~~ event is initiated by closing one of the MSIVs in a conservatively short time of 0.1 second. The event is bounded by TT for the RCS overpressure criterion because three SGs remain available to remove energy from the primary system. The event is analyzed in separate cases to evaluate overpressure in the affected SG and compliance with SAFDLs, particularly DNB. To obtain a conservative response for these cases, the analysis does not credit the non-safety-related turbine bypass system. The worst single failure is the failure of the MSRT to open in the affected loop.

The peak secondary pressure case is biased to maximize the heat transfer from the RCS to the secondary system. This heat transfer is maximized by assuming zero percent SG tube plugging and by assuming that a LOOP does not occur. In addition, the non-safety-related PZR spray is simulated to reduce the potential for RT on high PZR pressure.

The MDNBR case is biased to minimize the heat transfer from the RCS to the secondary system. This heat transfer is minimized by assuming five percent SG tube plugging and by assuming a LOOP at the time of TT. In addition, the non-safety-related PZR spray is simulated to reduce the increase in RCS pressure, which acts to reduce DNB margin. The analysis conservatively accounts for the effect of asymmetric core inlet coolant temperatures on core reactivity and power distributions.

Table 15.2-4—MSIVC Secondary Overpressurization - Key Input Parameters presents a listing of the key initial inputs used in this analysis. Table 15.2-5—MSIVC Overpressurization - Key Equipment Status presents a listing of the status of key systems.

15.2.4.3 Results

15.2.4.3.1 Peak Secondary Pressure Analysis Results

Table 15.2-6—MSIVC Secondary Overpressurization - Sequence of Events presents the sequence of events for the MSIVC maximum secondary pressure analysis. The MSIV in one loop is assumed to close in 0.1 second. The loss of steam flow from the affected SG causes an increase in demand on the unaffected SGs (Figure 15.2-14—MSIVC Secondary Overpressurization—MSIV Flow Rates). The affected SG pressurizes while the unaffected SGs depressurize (Figure 15.2-23 MSIVC Secondary Overpressurization—Maximum Secondary Pressure).

Pressure in the affected SG reaches the high SG pressure RT setpoint at 5.54 seconds initiating RT. This PS signal also opens the MSRIV in the affected SG, but the main steam relief control valve (MSRCV) is assumed to be failed in the closed position (although normally full open during hot full power operation). Steam flows through the MSRIV just long enough to pressurize the relief train piping ahead of the closed MSRCV. This result is shown in Figure 15.2-15—MSIVC Secondary Overpressurization - Safety Valve Flows for the Affected Loop.

The depressurization of the three unaffected SGs causes an increase in the primary to secondary heat transfer for those loops. This condition lowers the temperature of the fluid returning to the core from the unaffected loops (Figure 15.2-16—MSIVC Secondary Overpressurization—Cold Leg Temperatures). The cooler water returning to the core combined with a negative end of cycle moderator coefficient causes an increase in power (Figure 15.2-19—MSIVC Secondary Overpressurization — Reactor Power).

The pressure in the affected SG continues to increase until the first MSSV opens at 9.05 seconds (Figure 15.2-15). The opening setpoints and capacity of the MSSVs are adequate to limit peak secondary pressure (at the bottom of the SGs) to 1541 psia at 12.13 s, which is less than the acceptance criterion of 110 percent of the secondary system design pressure (1593.2 psia).

15.2.4.3.2 DNBR Analysis Results

15.02.01 - 15.02.05-8

The S-RELAP5 calculation indicates an increase in core power associated with the reduction in cold leg temperatures in the unaffected loops. This result is an artifact of use of these cold leg temperatures as input to the point kinetics model in S-RELAP5. ~~When evaluated realistically using 3-D kinetics, core power does not increase during the period prior to RT at approximately 6.44 s.~~ During the same time period prior to RT, RCS pressure increases slightly. The net result is that this event does not produce core conditions that challenge the MDNBR criterion.

– Response: Instrument and measurement uncertainties are conservatively applied for the single MSIVC analysis.

15.2.5 Steam Pressure Regulator Failure

The steam pressure regulator failure applies to BWR plants and is not applicable to the U.S. EPR.

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

The LNEP to the station auxiliaries is initiated by a complete loss of either the external (offsite) grid or the onsite AC distribution system. The complete loss of coolant flow event described in Section 15.3.2 is analyzed assuming a LOOP or LNEP is the initiating event. Refer to Section 15.3.2 for the short-term response of the LNEP event with respect to SAFDLs. The subsequent evolution of the event after RT is similar to and bounded by the LNFF scenario with a LOOP, which causes a complete loss of RCS flow; see Section 15.2.7 for the long-term plant response to an LNEP event.

15.02.06-1

Following initiation of the LNEP event, the diesel generators are started and provide electric power to vital loads. The sensible and decay heat loads are handled by the steam relief and EFW systems. Different segments of this event have similarities to the complete LNFF and complete loss of RCS flow events.

The loss of power results in immediate RCP coastdown and MFW termination. A RT occurs on low pump speed, if not sooner, as a result of the loss of power. Decaying reactor coolant flow causes an immediate increase in core coolant temperatures. Also, the sudden loss of subcooled MFW flow, the decaying reactor coolant flow, and termination of steam flow to the turbine all cause the SG heat removal rates to decrease. The decrease in SG heat removal rates augments the increase in reactor coolant temperatures.

As reactor coolant temperatures rise, the reactor coolant expands and surges into the PZR, potentially overfilling it. The resulting increase in RCS pressure causes the PZR PSRVs to open prior to termination of the short-term-heatup phase of the event that is terminated by reactor scram.

The termination of steam flow to the turbine and the continuing primary-to-secondary transfer of the decaying core power cause SG pressures to rapidly increase. The SGs are protected from overpressurization by opening of the MSRT and, if necessary by the MSSVs. The SG pressures stabilize with pressure controlled at the MSRT opening setpoint, with the steam release providing cooling of the RCS.

Liquid levels in the SGs soon decrease to the low-low setpoint, which actuates the EFW system and isolates the blowdown lines. EFW actuation and blowdown isolation occur for each SG independently. That is, when the low-low setpoint is reached in

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one SG, the EFW is actuated for only that SG, and only its blowdown line is isolated. When delivery of EFW begins, the rate of level decrease slows in the SGs fed by EFW, and the EFW provides additional cooling of the SGs and RCS. EFW supply to two of the SGs will be lost, one due to single failure and a second due to preventative maintenance.

In the long term, the two SGs that are not fed by EFW will dry out, thereby reducing the energy removal rate from the RCS. At 30 minutes into the transient, operator action can be credited for cross-connection of the EFW so that all four SGs are fed. Eventually, the decay heat level decreases below the level of heat removal via the MSRTs, marking the end of the challenge to the event acceptance criteria.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Event Description

The LNFF is an AOO in which there is a postulated complete termination of MFW. This condition can be caused by a LOOP or a malfunction in the MFW control system or equipment.

The sudden loss of subcooled MFW flow while the plant continues to operate at power causes SG heat removal rates to decrease, which causes reactor coolant temperatures to increase, thus expanding the RCS fluid. RCS fluid flows into the PZR, thereby increasing pressure, actuating the PZR spray system if available and potentially causing the PSRVs to open to control pressure. SG liquid levels drop steadily following termination of the MFW flow, quickly reaching the low SG narrow range (NR) level RT setpoint. This condition initiates RT and subsequent TT, thereby ending the short-term heatup phase of the event.

If available, the non-safety-related turbine bypass system opens to control secondary side pressure. If not available, SG pressure increases until the safety-related MSRTs open. SG levels continue to drop and soon reach the low SG level EFW setpoint that actuates EFW and isolates the SG blowdown line. When the delivery of EFW begins, the rate of level decrease in the fed SGs slows. For the SGs receiving EFW, liquid levels stabilize and begin to rise. If trains of EFW are unavailable due to maintenance or single failure, the operator can redirect the available EFW to feed the four SGs. The plant transitions a stable controlled state.

The LNFF event is classified as an AOO. The principal acceptance criteria that apply to this event are listed below:

- DNB SAFDL. The DNB acceptance criterion requires that minimum DNB ratio (MDNBR) is not less than the 95/95 correlation limit.

Figure 15.2-65—FWLB Representative Small Break – RCS Maximum Pressure

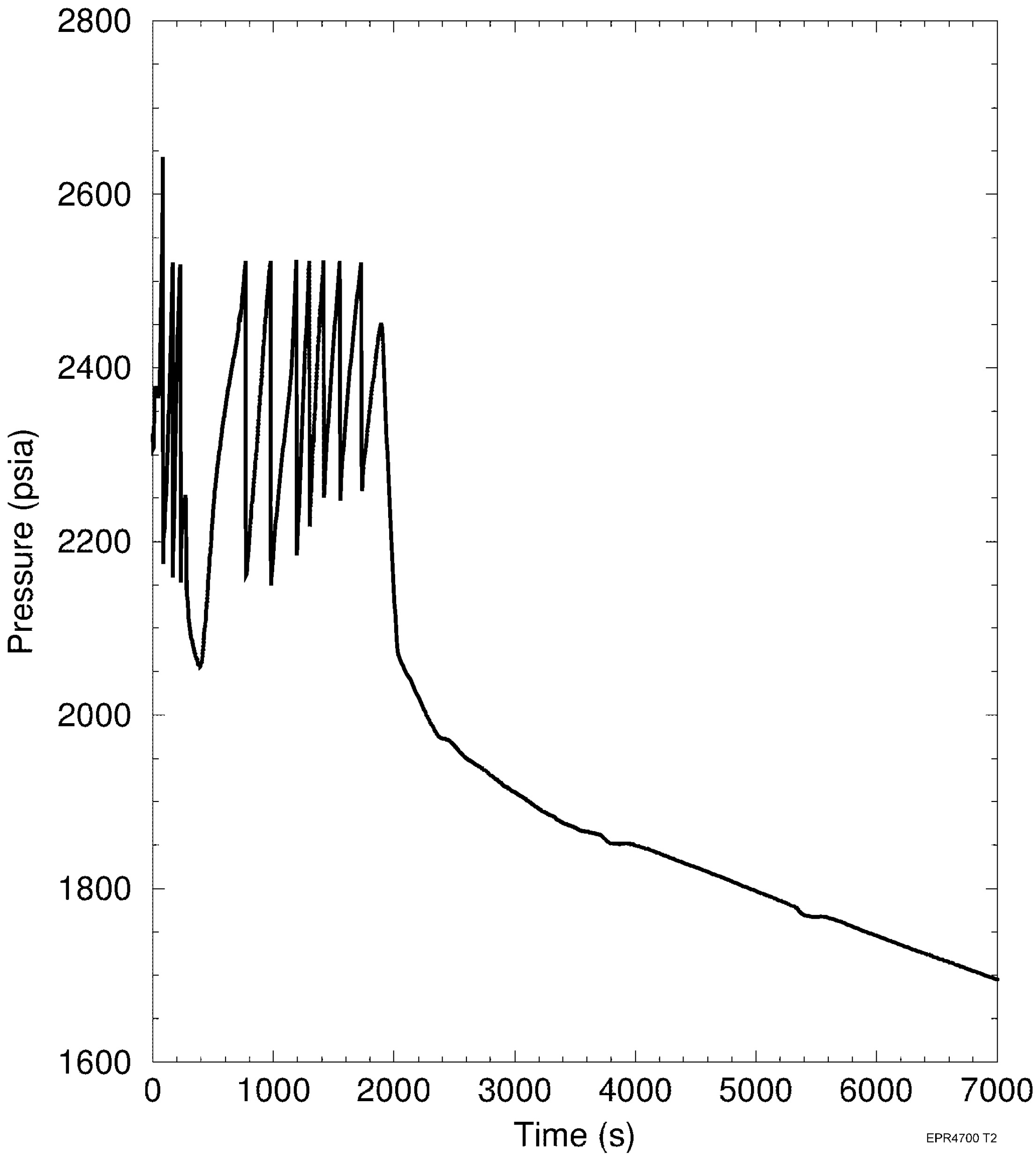

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Figure 15.2-66—FWLB Representative Small Break – Steam Generator Maximum Pressure

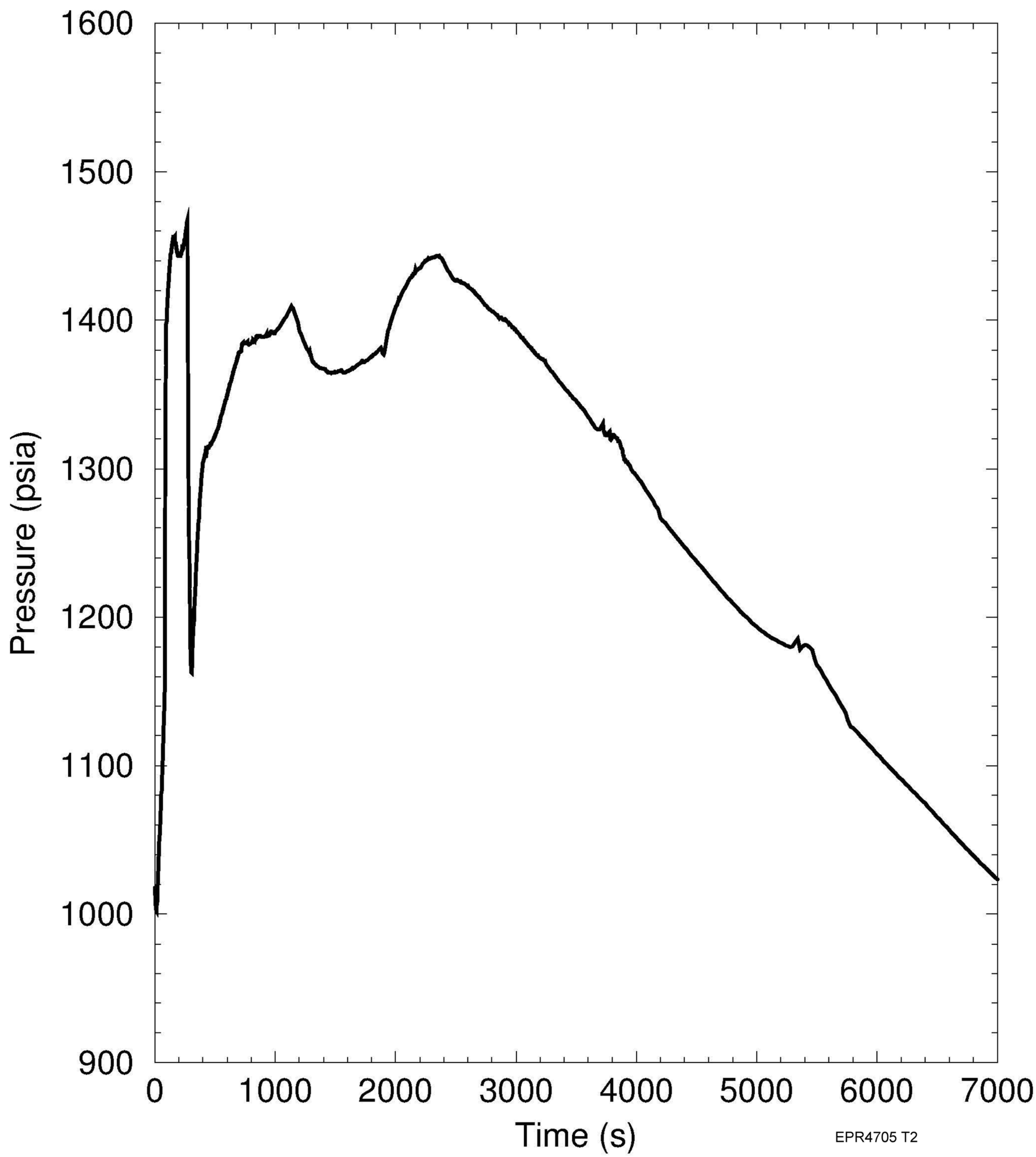

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Figure 15.2-67—FWLB Representative Small Break – Reactivities

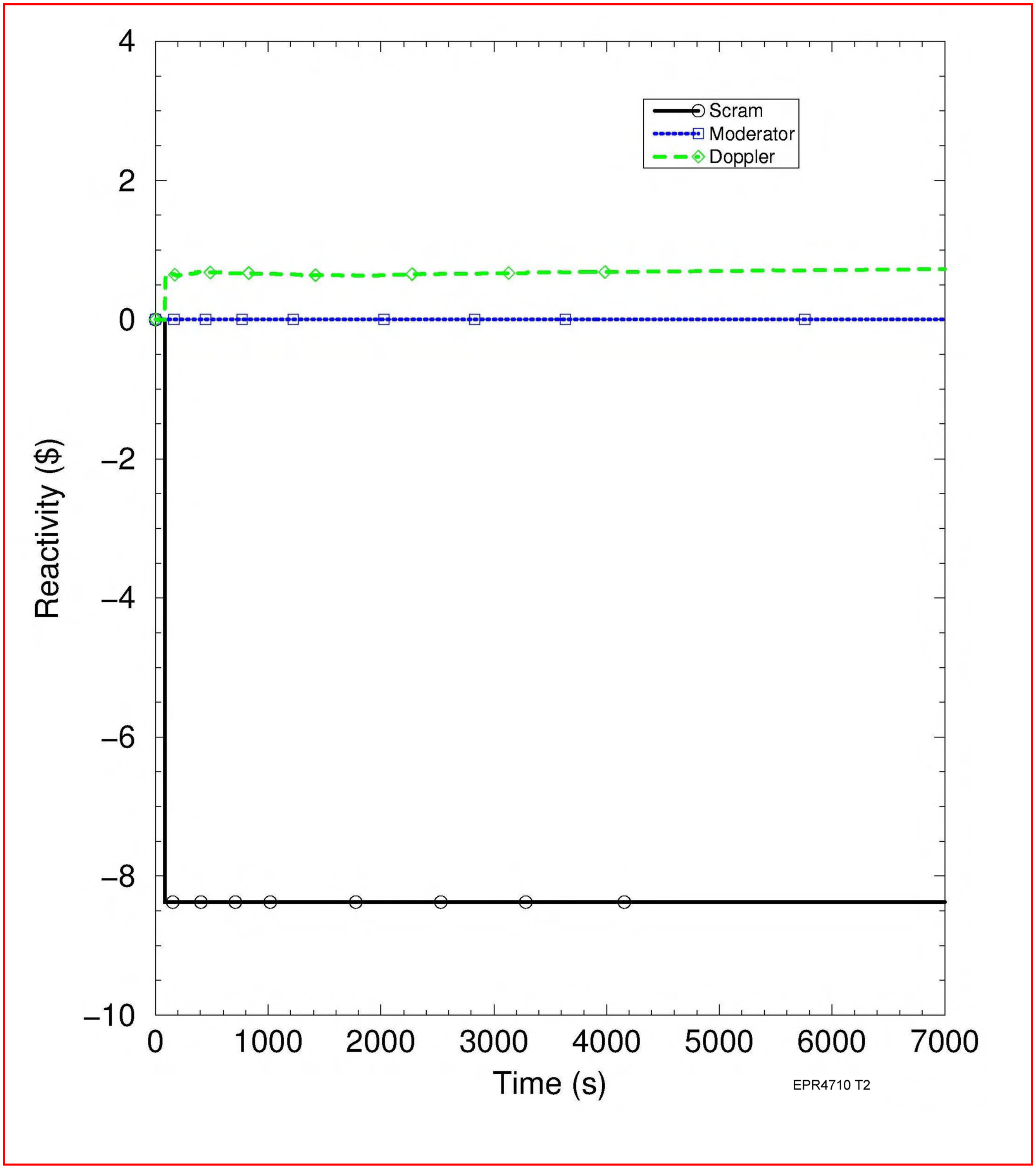

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Figure 15.2-68—FWLB Representative Small Break – Liquid Volume Fraction in Pressurizer Dome

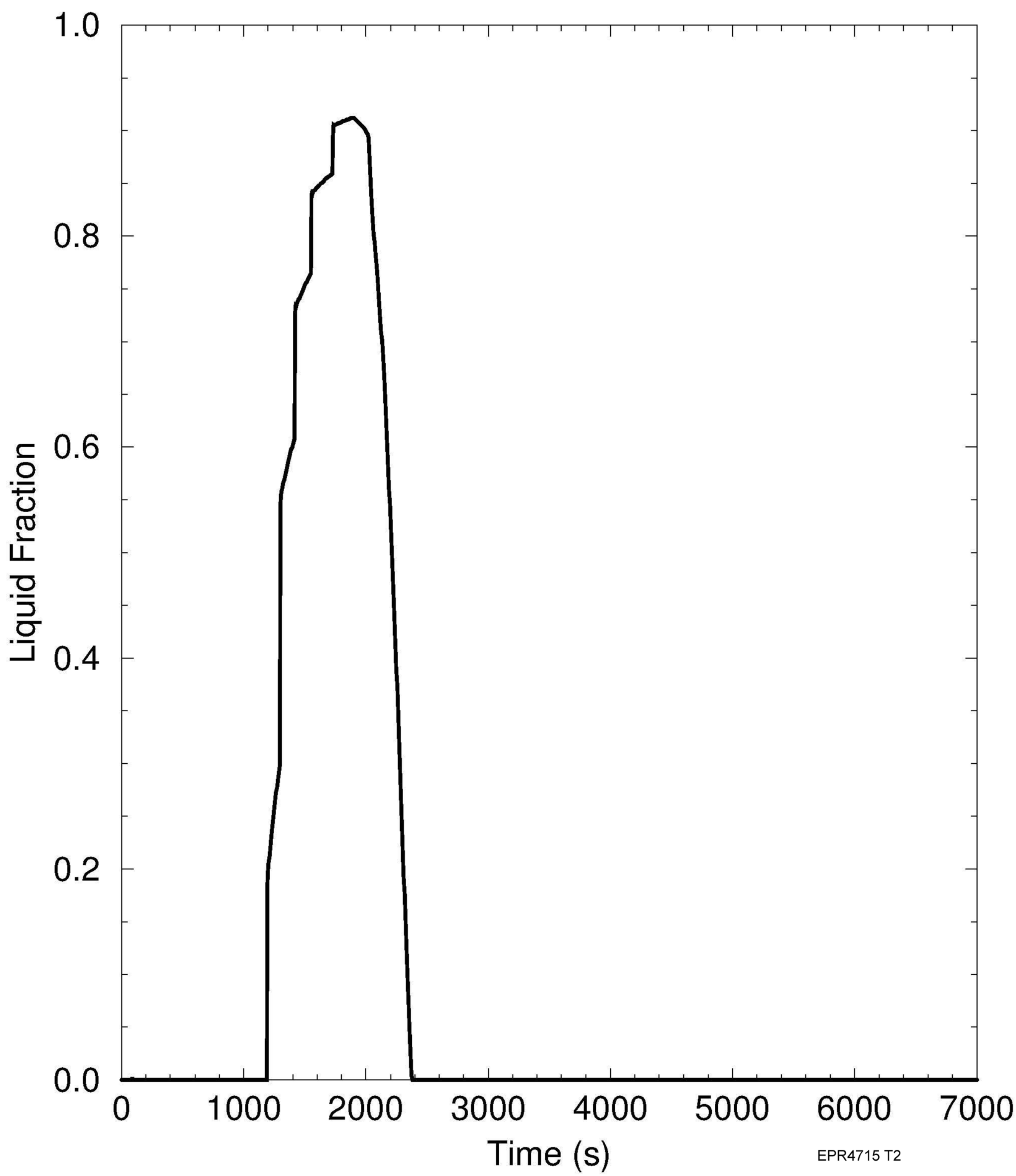

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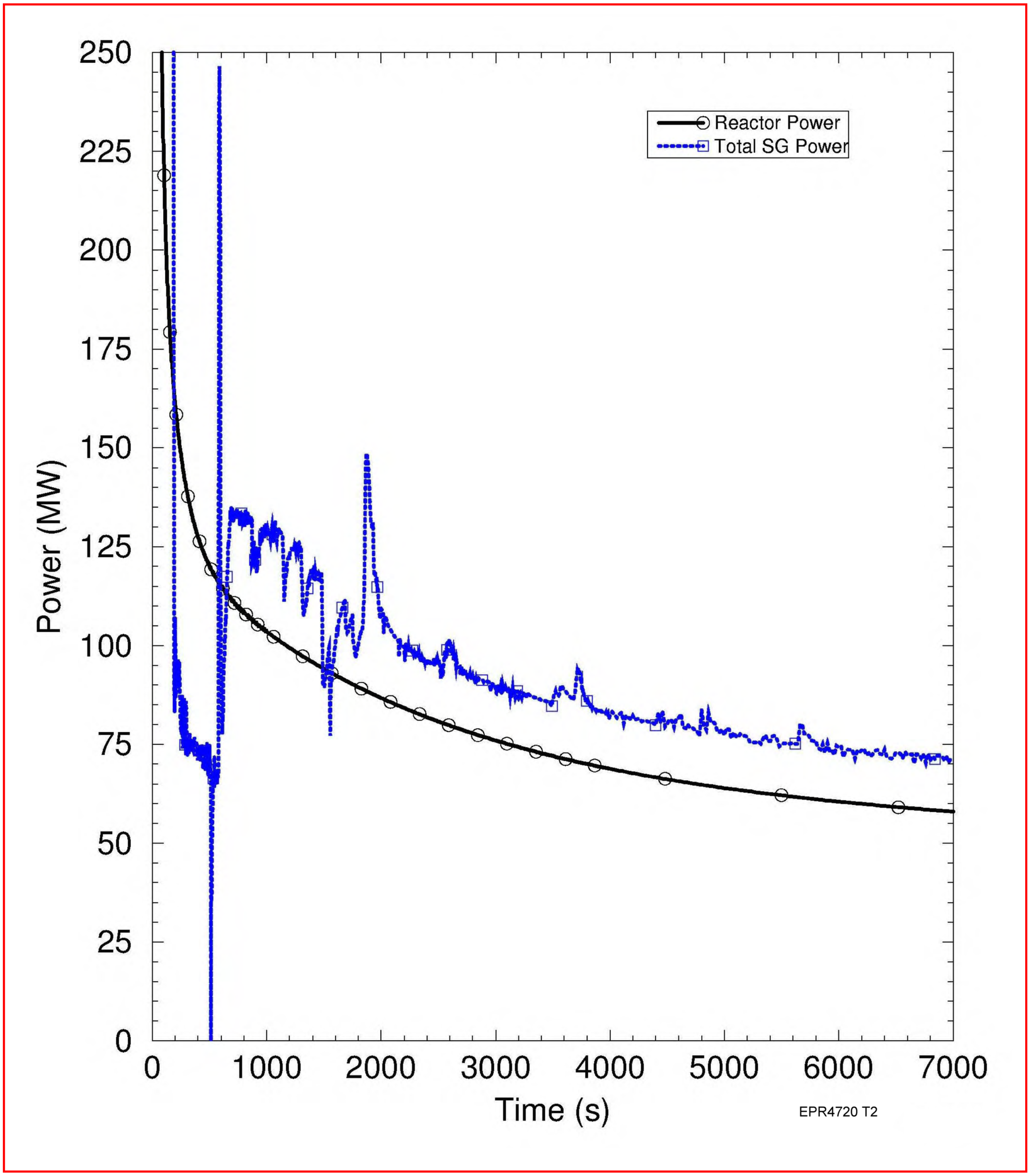
Figure 15.2-69—FWLB Maximum RCS Pressure Case – Reactor and Total Steam Generator Power



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- An AOO should not develop into a more serious plant condition without other faults occurring independently.

The focus for this event is meeting the specified acceptable fuel design limits (SAFDL).

15.6.1.2 Method of Analysis and Assumptions

The methodology used for this event analysis is described in the Codes and Methods Applicability Report for the U.S. EPR (Reference 1). It uses the S-RELAP5 computer code (described in Section 15.0.2) to calculate the transient thermal and hydraulic response of the primary and secondary systems. The code simulates the necessary components and has the properties necessary to model an IOPSRV event. The calculated transient boundary conditions for the reactor core from the S-RELAP5 analysis are used as input to the thermal margin calculations. The low DNB channel algorithm and the high LPD channel alogorithm are ~~is~~ simulated to predict RT and the adequacy of the dynamic compensation of the algorithm consistent with the Incore Trip Setpoint and Transient Methodology for U.S. EPR (Reference 2).

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Table 15.6-1—IOPSRV Event - Key Input Parameters presents the initial conditions for the limiting case. Table 15.6-2—IOPSRV Event - Key Equipment Status presents the status of mitigating equipment and components. The analysis begins at full power, under normal operating conditions. To minimize the heat removal by the secondary system, the maximum number of plugged SG tubes (five percent) is assumed.

The most reactive control rod is assumed not to insert at RT. LOOP is assumed to occur with RT. Subsequent to an RT, the limiting single failure is taken as the failure of one emergency diesel generator (EDG), resulting in the unavailability of one train of pumped SIS (MHSI, LHSI, and EFWS). A second EDG is assumed to be under maintenance and therefore unavailable, causing a second train of pumped SIS to be unavailable.

Degraded conditions are assumed for the MHSI pump startup and flow rates to produce the most conservative emergency core cooling system (ECCS) response. Degraded containment conditions are also assumed so that the actuation setpoints of mitigating systems use the largest instrument uncertainties.

Operator actions are credited at 30 minutes into the event to align EFWS flow from the two operational trains of EFWS to the four SGs. Later, operator actions are necessary to transition the plant from a controlled state to a safe shutdown condition.

The limiting case uses beginning-of-cycle (BOC) fuel conditions and assumes the rod position controller is in manual mode. At the BOC, the boron concentration is at its highest. A decrease in density following the IOPSRV results in a decrease in boron concentration. The resulting positive reactivity feedback causes a power increase in the early phase of the event.

End-of-cycle (EOC) fuel conditions are considered in a sensitivity calculation with the assumption the rod position controller is in automatic mode. At EOC, a decrease in density causes negative reactivity feedback because the boron concentration is lower. The rod position controller responds to the core average temperature and turbine generator demand. These parameters do not change rapidly. The net effect is that a decrease in reactor power occurs prior to reaching the RT signal, and this case is less limiting compared to the base BOC case.

Sensitivity studies were also conducted to bound uncertainties in PSRV flow rate (at 20 percent) and core decay heat (at 20 percent). These uncertainties are taken into account in the limiting case used for the thermal-hydraulic DNB analysis. Both uncertainties are included in the limiting case presented.

15.6.1.3 Results

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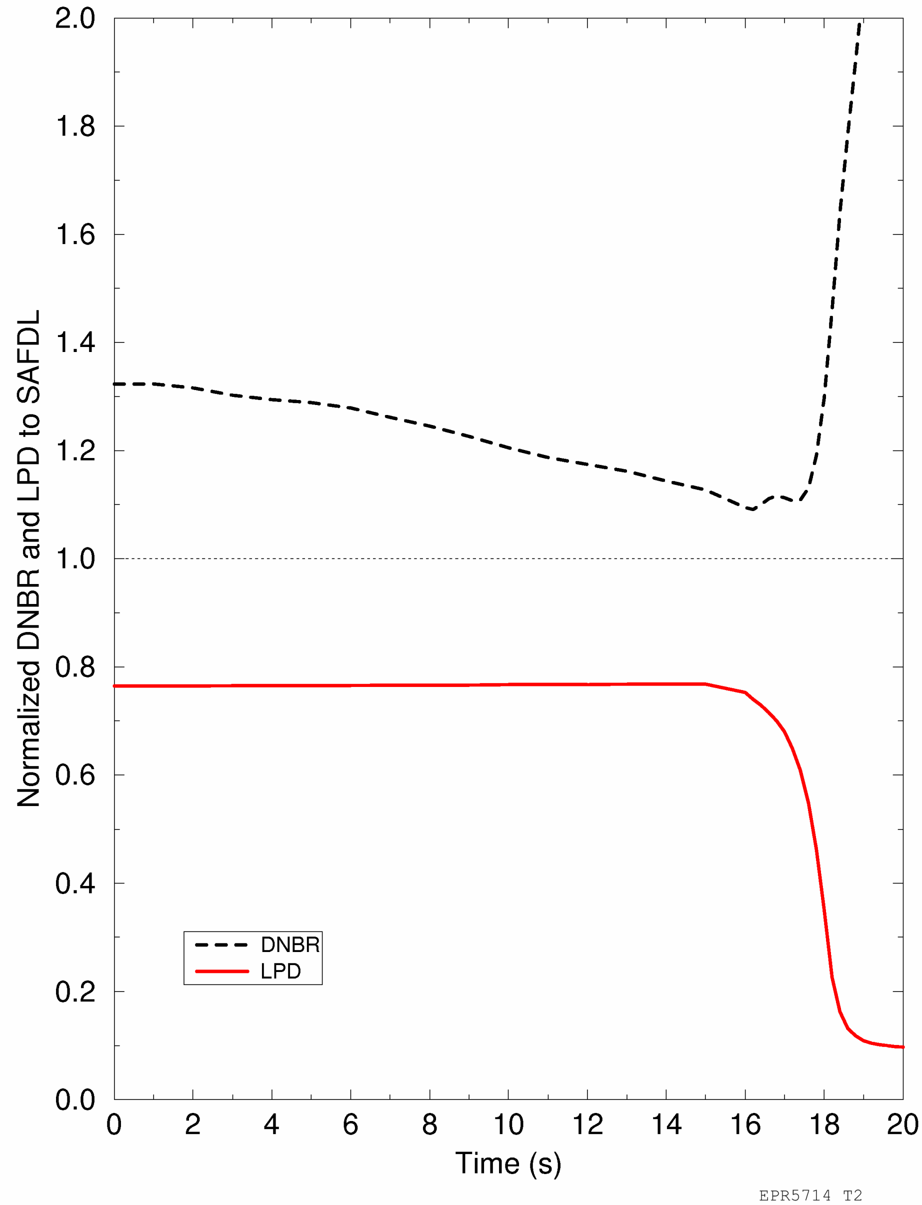
Table 15.6-3—IOPSRV Event - Sequence of Events presents the sequence of events for this case. Figure 15.6-7—IOPSRV Event - Pressurizer Levelpresents the PZR level after the PSRV opens. After the PSRV opens, reactor power increases slightly prior to RT at 39 seconds (Figure 15.6-1—IOPSRV Event - Transient Reactor Power). The increase in reactor power causes a small increase in core average heat flux (Figure 15.6-6—IOPSRV Event - Core Average Heat Flux). The primary pressure decreases throughout most of the event (Figure 15.6-2—IOPSRV Event - PZR Pressure). The core inlet temperature is stable prior to the RT (Figure 15.6-8—IOPSRV Event - Core Inlet Temperature). Figure 15.6-93—IOPSRV Event - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

~~The DNB RT setpoints and the dynamic compensation built into the low DNB channel algorithm are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL. The minimum DNB that can be reached under any condition is the SAFDL from the ACH-2 CHF Correlation for the U.S. EPR (Reference 3). The maximum linear power density (LPD) realized during the IOPSRV event is below the limit. Therefore, the peak fuel centerline temperature remains below the fuel melting point.~~The DNB reactor trip (RT) and high LPD RT setpoints, as well as the dynamic compensation built into the low DNBR channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, the DNB limiting condition for operation (LCO) and the LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel the fuel cladding integrity is maintained and the peak centerline temperatures remain below the fuel centerline melt limit. Figure 15.6-8 through Figure 15.6-10 show the effect of the partial cooldown initiated by the low PZR pressure SI signal. The controlled decrease

Figure 15.6-93—IOPSRV - Representative Plot of Minimum Normalized Minimum DNBR and Maximum LPD Normalized to the SAFDL



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