Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

December 29, 2011

L-2011-533
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Plant Unit 2
Docket No. 50-389
Renewed Facility Operating License No. NPF-16

Response to NRC Reactor System Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-021), "License Amendment Request for Extended Power Uprate," February 25, 2011, Accession No. ML110730116.

(2) Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 2 EPU – draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB)," September 6, 2011.

(3) Email from L. Abbott (FPL) to T. Orf (NRC), "Re: St. Lucie 2 EPU – draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question Numbering," September 28, 2011.

By letter L-2011-021 dated February 25, 2011 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. NPF-16 and revise the St. Lucie Unit 2 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an extended power uprate (EPU).

A001
NRR

In an email dated September 6, 2011 from NRC (T. Orf) to FPL (C. Wasik) [Reference 2], the NRC staff requested additional information regarding FPL's license amendment request (LAR) to implement the EPU. FPL email dated September 28, 2011 from FPL (L. Abbott) to NRC (T. Orf) [Reference 3], provided specific numbers (SXRB-01 through SRXB-102) for the questions included in the September 6, 2011 email. The attachment to this letter provides the FPL responses to RAI questions SRXB-78 thorough SRXB-95, SRXB-97, and SRXB-98 related to boric acid precipitation and loss of coolant accident (LOCA) analyses. The remaining responses, including SRXB-96, are being provided in separate submittals.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2011-021 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-467-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on 29 - December - 2011

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment

cc: Mr. William Passetti, Florida Department of Health

Response to Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information

The following information is provided by Florida Power & Light (FPL) in response to the U.S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support the review of Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Nuclear Plant Unit 2 that was submitted to the NRC by FPL via letter (L-2011-021), February 25, 2011, Accession No. ML110730116.

In an email dated September 6, 2011 from NRC (T. Orf) to FPL (C. Wasik), "St. Lucie 2 EPU – draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB)," the NRC staff requested additional information regarding FPL's request to implement the EPU. FPL email dated September 28, 2011 from FPL (L. Abbott) to NRC (T. Orf), "Re: St. Lucie 2 EPU – draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question Numbering," provided specific numbers (SXRB-01 through SRXB-102) for the questions included in the RAI. The responses to RAI questions SRXB-78 through SRXB-95, SRXB-97, and SRXB-98 are provided below. The remaining responses, including SRXB-96, are being provided in separate submittals.

IV. Boron Acid Precipitation and LOCA Analyses (Attachment 5 to Licensing Report)

SRXB-78 (RAI 2.8.5.6.3-1)

Please provide the following information for the St. Lucie Unit 2 NSSS:

a. Volume of the lower plenum, core and upper plenum below the bottom elevation of the hot leg, each identified separately. Also provide heights of these regions.

b. Loop friction and geometry pressure losses from the core exit through the steam generators to the inlet nozzle of the reactor vessel. Also, provide the locked rotor RCP k-factor. Please provide the mass flow rates, flow areas, k-factors, and coolant temperatures for the pressure losses provided (upper plenum, hot legs, SGs, suction legs, RCPs, and discharge legs). Please include the reduced SG flow areas due to plugged tubes. Please also provide the equivalent loss coefficient through the loop to a break in the single broken cold leg. Also identify the flow area (hydraulic diameter) the k-factors are based on.

Response

a. The volumes and heights of the lower plenum, core, and upper plenum are documented in Table SRXB-78-1 below.

Table SRXB-78-1
Reactor Coolant System
Volumes, Areas, and Elevation / Heights

Parameter	Value (units)
Lower Plenum	
Height of the lower plenum	9.715 ft
Volume of the lower plenum	827.7 ft^3
Active Core (actual, i.e., no voids)	
Height of the active core	11.392 ft
Area of the active core	54.82 ft^2
Volume of the active core	624.51 ft^3
Outlet Plenum (top of active core to top of the hot leg)	
Height of the outlet plenum	7.962 ft
Volume of the outlet plenum	781.5 ft^3

b. The loop friction and geometry pressure losses are documented below in Tables SRXB-78-2 and SRXB-78-3, along with Figures SRXB-78-1 and SRXB-78-2, which are provided for informational purposes. Additionally, Table SRXB-78-2 and Figure SRXB-78-2 provide information on the reactor vessel, which was not specifically requested, but is provided for completeness.

Table SRXB-78-2
Calculation of Station-to-Station Reactor Vessel K-Factors

Stations	Pressure Drop (psi)	Specific Volume (ft^3/lbm)	Flow Rate (lbm/hr)	Kg* (Acore)	Kf* (Acore)
1-2	0.60	0.0214865	1.3998E+08	0.514	
90°	5.15	0.0214865	1.3998E+08	4.414	
2-3	0.04	0.0214865	1.3998E+08	0.034	
3-4	0.52	0.0214865	1.3998E+08	0.446	
4-5	0.49	0.0214865	1.3998E+08	0.420	
5-6	0.00	0.0214865	1.3998E+08	0.000	
6-7 (friction)	0.60	0.0214865	1.3998E+08		0.514
7-8	0.32	0.0214865	1.3998E+08	0.274	
8-S	5.21	0.0214865	1.3998E+08	4.466	
S-9	0.10	0.0214865	1.3998E+08	0.086	
9-11	1.96	0.0214865	1.3998E+08	1.680	
11-13	0.36	0.0214865	1.3998E+08	0.309	
13-15	2.09	0.0214865	1.3998E+08	1.791	
15-17	1.66	0.0214865	1.3998E+08	1.423	
17-a (friction)	5.49	0.0224454	1.3998E+08		4.505
17-a (geometry)	7.78	0.0224454	1.3998E+08	6.384	
a-18 (friction)	0.32	0.0224454	1.3998E+08		0.263
a-18 (geometry)	1.15	0.0224454	1.3998E+08	0.944	
18-20	0.77	0.0236744	1.3998E+08	0.599	
20-24	7.93	0.0236744	1.3998E+08	6.169	
Total	42.54				

Note

* 10% uncertainty has been applied to the pressure drop values to calculate K-factors.

- Kg (geometric losses)
- Kf (frictional losses).

Additionally, the specific volumes provided in Table SRXB-78-2 above are based on the cold and hot leg coolant temperatures at EPU conditions. The flow area used in these calculations is the core flow area (Acore = 54.82 ft^2).

Table SRXB-78-3
Calculation of Station-to-Station K-Factors

Station	Geometry ΔP Forward Flow (psi)	Friction ΔP (psi)	Specific Volume (ft^3/lbm)	Flow Rate (lbm/sec)	Kg (Acore)	Kf (Acore)
1-2	0.00	0.18	0.0236744	19442	0.000	0.560
2-3	0.80	0.18	0.0236744	19442	2.489	0.560
3-4	5.20	0.00	0.0236744	19442	16.181	0.000
4-5	0.12	0.00	0.0236744	19442	0.373	0.000
5-6	0.15	39.24	0.0224454	19442	0.492	128.794
6-7	1.58	0.00	0.0214865	19442	5.417	0.000
7-8	4.60	0.00	0.0214865	9721	63.088	0.000
8-9	1.95	0.35	0.0214865	9721	26.744	4.800
9-10	1.36	0.35	0.0214865	9721	18.652	4.800
11-12	0.00	0.26	0.0214865	9721	0.000	3.566
12-13	1.09	0.26	0.0214865	9721	14.949	3.566

Note

* 10% uncertainty has been applied to the pressure drop values to calculate Kf (frictional losses) and 20% uncertainty has been applied to the pressure drop values to calculate Kg (geometric losses).

Additionally, the specific volumes provided in Table SRXB-78-3 above are based on the cold and hot leg coolant temperatures at EPU conditions. The flow area used in these calculations is the core flow area (Acore = 54.82 ft^2).

The reactor coolant pump locked rotor K-factor, based on the core flow area, is 2131.58 ft^2, where 17.03 is the locked rotor K-factor and 4.9 ft^2 is the area that the locked rotor K-factor is based on.

Figure SRXB-78-1
Reactor Coolant System Loop Sections

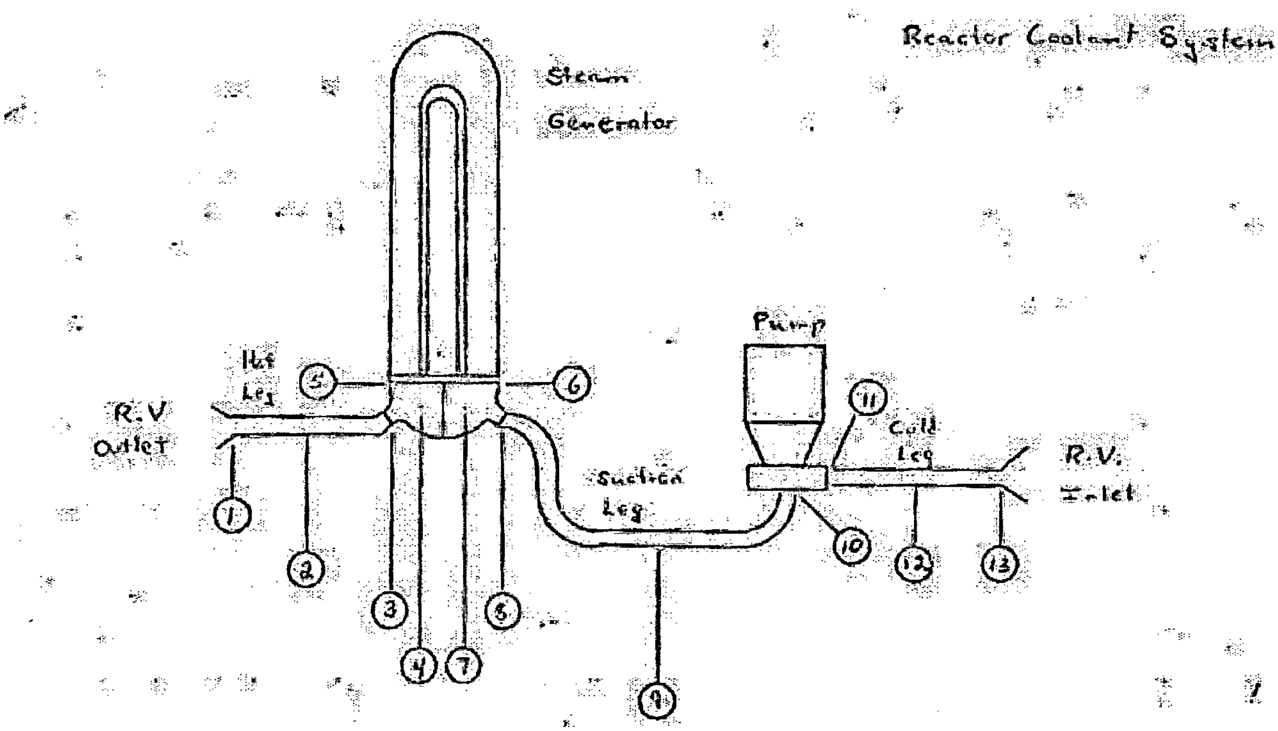
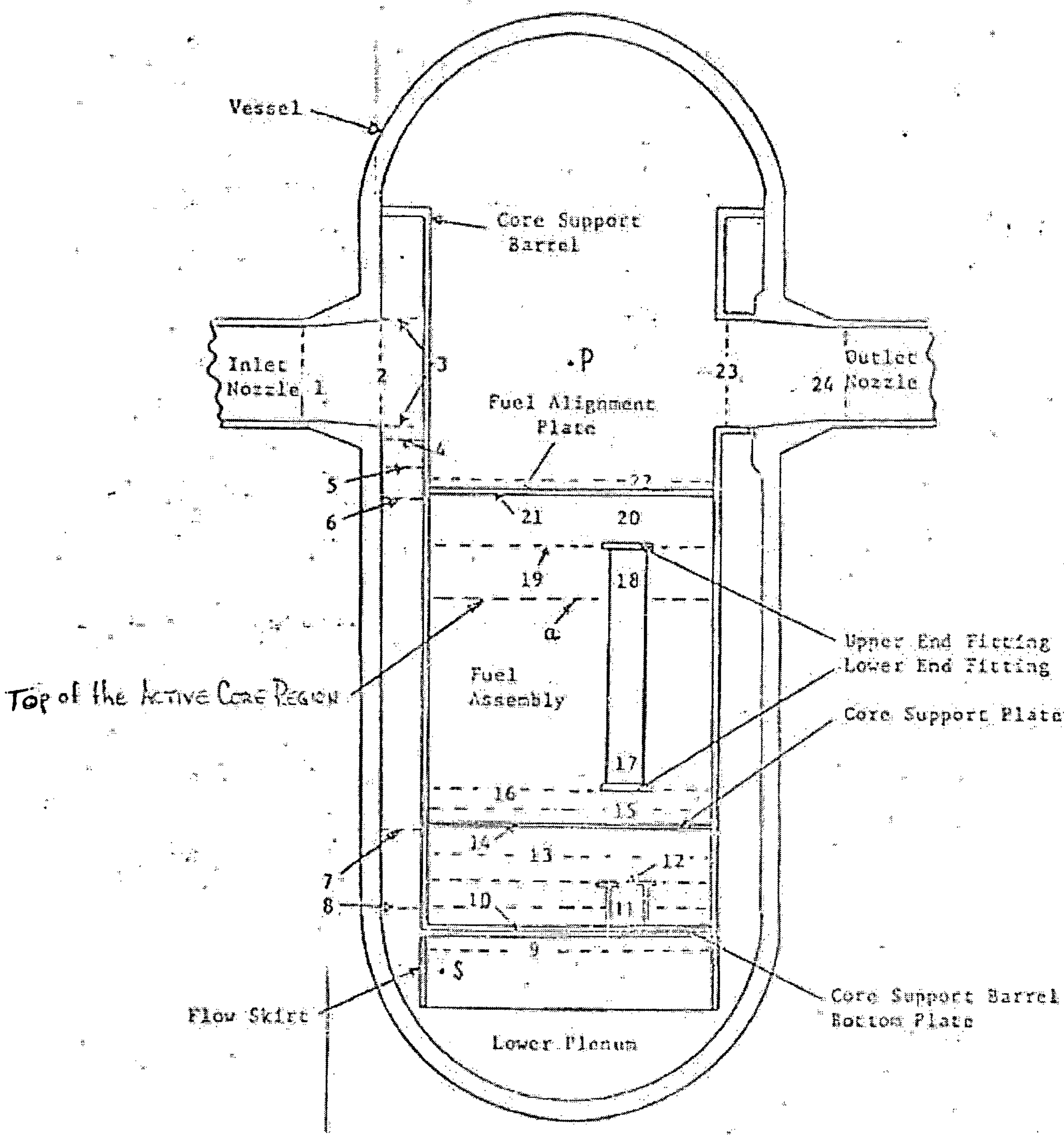


Figure SRXB-78-2
Reactor Vessel Stations



<u>SRXB-79 (RAI 2.8.5.6.3-2)</u>

What is the sump temperature versus time following recirculation and how does this impact precipitation? Is the boric acid concentration in the vessel below the precipitation limit based on the minimum sump temperature at the time the switch to simultaneous injection is performed? Please explain. What is the minimum temperature in the lower plenum just prior to recirculation actuation? Also please explain how the sump boric acid concentration is calculated and justify the values in Fig. 2.8.5.6.3-76?

<u>Response</u>

The BORON analysis assumes a constant core steam enthalpy and core inlet liquid enthalpy based on saturation conditions at 14.7 psia. Therefore, the analysis assumes a constant transient sump temperature, specifically 212°F. Containment temperature and pressure will be at or above atmospheric conditions, so modeling at atmospheric represents a conservative lower limit. The precipitation limit is also based on a saturation pressure of 14.7 psia, which corresponds to a temperature of 212°F. The current calculation of the precipitation limit is conservative, as it is based on a conservative pressure of 14.7 psia. At 6 hours, the maximum time for switchover to simultaneous injection, the boric acid concentration in the vessel is 26.3 wt%, which is less than the precipitation limit of 27.6 wt% based on the sump temperature of 212°F (14.7 psia). The minimum temperature in the lower plenum cannot be lower than the minimum temperature in the sump; therefore, modeling the lower plenum temperature as 212°F (temperature at atmospheric conditions) is conservative.

The sump boric acid concentration values shown in EPU LAR Attachment 5, Figure 2.8.5.6.3-76 were calculated as part of the sump dilution analysis, i.e., the calculation of the minimum sump boron concentration following a large break loss of coolant accident (LOCA) as a function of initial reactor coolant system (RCS) boron concentration. The purpose of this analysis is to calculate the minimum boric acid concentration in the containment sump at the time of switchover to simultaneous hot/cold leg injection. This is important, as the potential exists for core recriticality following switchover to simultaneous hot/cold side injection. The results of this analysis are used as input to ensure that the containment sump boron concentration remains greater than the post-LOCA critical boron concentration to the time of emergency core cooling system (ECCS) switchover to hot/cold leg recirculation. The concentration was calculated using the BORON code that is used for the boric acid precipitation analysis, with minimum tank liquid volumes and minimum tank boron concentrations, to minimize the sump boron concentration.

SRXB-80 (RAI 2.8.5.6.3-3)

The mixing volume in the inner vessel includes the mixture level in the upper plenum to the top of the hot leg. Since vapor generated in the core exits the vessel through the hot legs, the mixture level cannot rise to the top of the hot leg, particularly since the upper plenum pressure will increase sufficiently to depress the two-phase level low enough below the hot leg top elevation to allow the steam to vent through the loop. The mixture volume is expected to decrease to 0.5 to One foot below the top of the hot leg to allow the vapor to vent. What is the impact on precipitation time assuming the mixture level remains at lease one foot below the top of the hot leg? Please also clarify that only the volume in the upper plenum to this evaluation is credited in the mixing volume.

Response

The current analysis credits the volume in the upper plenum to an elevation to the top of the hot leg for the mixing volume calculation. To evaluate the impact of crediting a mixing volume height of one foot below the top of the hot leg, the mixing volume calculation was redone, reducing the height of the outlet plenum by one foot. Crediting the volume in the upper plenum to only the elevation specified above resulted in about a 4% reduction in the mixing volume. This has a small effect on the results of the boric acid precipitation analysis. Specifically, with no hot side injection, the time to reach the solubility limit of 27.6 wt% has decreased from 7.1 hours to 6.7 hours. Additionally, slight changes were seen in the results of the case when hot side injection of 250 gpm is initiated 6 hours post-loss of coolant accident (post-LOCA). These changes are documented below. However, the overall results remain applicable. Beginning simultaneous hot and cold side injection between 4-6 hours post-LOCA with a hot side injection of 250 gpm continues to provide acceptable results that preclude boric acid precipitation.

Description	Hot Side Injection Start Time (hr)	Hot Side Injection Flow Rate (gpm)	Max. Core Boric Acid Concentration (wt%)	Margin to Solubility Limit (wt%)	Time of Maximum Concentration (hr)
Original mixing volume (7800 gallons)	6.0	250	26.3	1.3	7.3
New reduced mixing volume (7480 gallons)	6.0	250	27.3	0.3	7.3

SRXB-81 (RAI 2.8.5.6.3-4)

Is the mixing volume assumed to be a fixed volume in the method? Since decay heat is higher early in the event, the loop pressure drop will be higher and limit the growth of the mixing volume with time from the initiation of the LOCA. As such, the mixing volume cannot be fixed (constant) parameter until hot side injection is aligned. Please explain. Also, please show the mixture height vs time and the core and upper plenum void distributions at one hr intervals until the switch to hot side injection is made?

Response

The boric acid precipitation analysis assumes a fixed mixing volume. The mixing volume was calculated throughout the window for starting simultaneous hot and cold side injection (i.e., at 4, 5, and 6 hours) and it was found that the volume does not vary greatly. Additionally, boric acid precipitation in the core is more of a concern at later times (greater than 6 hours), and the use of a slightly larger mixing volume when the boric acid concentration in the core (at 4 and 5 hours) is lower, will have negligible impact on the final results. In other words, the later end of the window is still the limiting time for boric acid precipitation concerns.

Regardless of time, the height of the outlet plenum credited for the mixing volume is assumed to be constant and credited to the top of the core support barrel nozzle (the outlet plenum height is 7.9620 feet). Table SRXB-81-1 documents the core and upper plenum void distributions for 1-6 hours.

SRXB-81-1
Void Fractions

Mixing Volume Region*	Height (feet)	Time (hours)					
		1	2	3	4	5	6
1	9.7150	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
2	1.1392	0.4621	0.4194	0.3949	0.3778	0.3649	0.3543
3	1.1392	0.6321	0.5910	0.5662	0.5485	0.5347	0.5232
4	1.1392	0.7204	0.6843	0.6619	0.6456	0.6329	0.6221
5	1.1392	0.7746	0.7429	0.7230	0.7084	0.6968	0.6870
6	1.1392	0.8111	0.7832	0.7654	0.7523	0.7418	0.7329
7	1.1392	0.8375	0.8125	0.7966	0.7847	0.7752	0.7670
8	1.1392	0.8574	0.8349	0.8204	0.8096	0.8009	0.7934
9	1.1392	0.8730	0.8525	0.8393	0.8293	0.8213	0.8145
10	1.1392	0.8855	0.8667	0.8545	0.8453	0.8380	0.8316
11	1.1392	0.8957	0.8784	0.8671	0.8586	0.8518	0.8459
12	7.9620	0.8275	0.8014	0.7847	0.7723	0.7624	0.7540

Note

* Mixing Volume Region 1 is the lower plenum and Region 12 is the outer plenum.

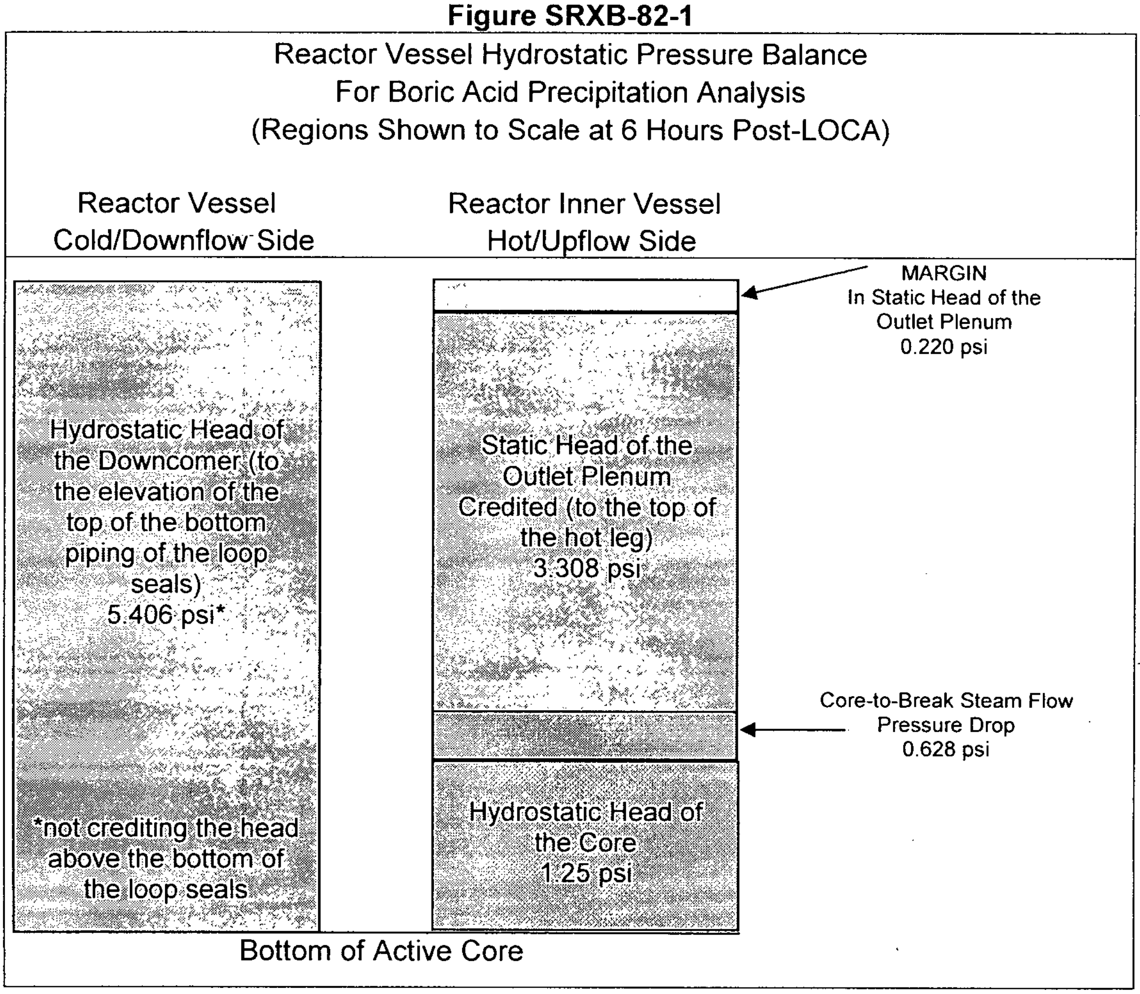
SRXB-82 (RAI 2.8.5.6.3-5)

What is the impact on precipitation timing for breaks on the top of the cold leg with the loop seal region assumed filled with liquid with the core steaming rate bubbling through the vertical section at the pump suction piping? Show the boric acid concentration vs time assuming the break is on the top of the discharge leg and vapor bubbles through the loop seals. Also show the loop seal void fraction vs time and the method used to compute the loop seal void fraction. Please explain.

Response

An evaluation was performed to demonstrate that the limiting break assumed in the boric acid precipitation analysis, a double-ended break in the cold leg, is in fact the most limiting. A description of this evaluation has been incorporated in EPU LAR Attachment 5, Section 2.8.5.6.3.5, which is based on a calculation of the reactor vessel hydrostatic pressure balance to identify the margin that is available to overcome any additional pressure due to loop seal clearing. Figure SRXB-82-1 summarizes these calculations.

Figure SRXB-82-1



The discussion from EPU LAR Attachment 5, Section 2.8.5.6.3.5.2, of how loop seal refilling and clearing is accounted for, has been included below, to provide a full response to this RAI, and is as follows:

The above Figure SRXB-82-1 represents a slot break at the top of the cold leg. This includes the potential additional pressure drop due to the loop seals refilling, as well as, an additional pressure drop due to a higher level for the downcomer liquid, described as follows:

- Loop seal refilling will increase the value of the core-to-break steam flow pressure drop, which will reduce the margin calculated above.
- Note that the hydrostatic head above the bottom of the cold legs does not need to be included since it balances on the downcomer and loop seal side of the hydrostatic pressure balance.

To demonstrate that a slot break at the top of the cold leg is capable of clearing the loop seals, the loop seals hydrostatic head is calculated and deducted from the hydrostatic head of the downcomer. For the pressure balance to be acceptable, either the pressure must be equal, or it must be shown that there is still margin in the static head of the outlet plenum.

Using the geometric information tabulated in EPU LAR Attachment 5, Table 2.8.5.6.3-6, the height of the loop seal (from top of cross-over leg to bottom of discharge leg) is 3.5 ft. The static head associated with the height of liquid in the cold leg above the loop seal inlet to the reactor coolant pump is offset by the added static head for the downcomer from this liquid. The pressure drop, ignoring the head of steam in the downflow side of the loop seal, associated with clearing the liquid in the upflow side of the loop seal is calculated as follows:

$$\Delta P_{DC} = 3.5 \text{ ft} / 0.016714 \text{ ft}^3/\text{lbm} / 144 \text{ in.}^2/\text{ft}^2 * g/g_c = 1.454 \text{ psi}$$

This pressure drop has been deducted from the hydrostatic head of the downcomer. Constructing a reactor vessel hydrostatic pressure balance based on this, there remains 0.220 psi available margin in the pressure balance for the break in the top of the cold leg. Therefore, the pressure balance is acceptable, and the loop seals can be cleared. Note that the above calculation was done assuming zero void fraction in the loop seals. As noted in the RAI, steam would bubble through the vertical section of the loop seal. This will decrease the hydrostatic head of the loop seals and provide additional margin.

Other potential break locations are even less limiting and do not need to be evaluated. For example, if the break was in the loop seal, the above calculation of the pressure drop due to loop seal refilling is not required. Comparatively, the double ended break in the cold leg is still bounding.

Therefore, the boron precipitation analysis will not be affected by the described phenomena of the refilling of loop seals. This includes the calculation of precipitation timing, which remains bounding and unchanged.

Table SRXB-82-1 documents the boric acid concentration vs. time for a break on the top of the discharge leg. This case documents a hot side injection flow rate of 250 gpm beginning at 6 hours post-LOCA.

Table SRXB-82-1
Core Boric Acid Concentration vs. Time

Time (hours)	Core Concentration (wt%)		Time (hours)	Core Concentration (wt%)		Time (hours)	Core Concentration (wt%)
0.3	3.7		5.9	24.3		9.1	24.0
1.6	10.6		6.0	24.6		9.2	23.8
2.3	13.3		6.1	24.8		9.3	23.6
3.0	15.7		6.2	25.1		9.4	23.3
3.1	16.0		6.3	25.3		9.5	23.1
3.2	16.4		6.4	25.5		9.6	22.8
3.3	16.7		6.5	25.6		9.7	22.6
3.4	17.0		6.6	25.8		9.8	22.3
3.5	17.3		6.7	25.9		9.9	22.1
3.6	17.6		6.8	26.0		10.0	21.8
3.7	18.0		6.9	26.1		10.1	21.5
3.8	18.3		7.0	26.2		10.2	21.3
3.9	18.6		7.1	26.2		10.3	21.0
4.0	18.9		7.2	26.3		10.4	20.7
4.1	19.2		7.3	26.3		10.5	20.5
4.2	19.5		7.4	26.3		10.6	20.2
4.3	19.8		7.5	26.3		10.7	19.9
4.4	20.1		7.6	26.2		10.8	19.6
4.5	20.4		7.7	26.2		10.9	19.4
4.6	20.7		7.8	26.1		11.0	19.1
4.7	21.0		7.9	26.0		11.1	18.8
4.8	21.3		8.0	25.9		11.2	18.6
4.9	21.5		8.1	25.8		11.3	18.3
5.0	21.8		8.2	25.7		11.4	18.0
5.1	22.1		8.3	25.5		11.5	17.8
5.2	22.4		8.4	25.4		11.6	17.5
5.3	22.7		8.5	25.2		11.7	17.2
5.4	23.0		8.6	25.0		11.8	17.0
5.5	23.2		8.7	24.9		11.9	16.7
5.6	23.5		8.8	24.7		12.0	16.5
5.7	23.8		8.9	24.5		12.1	16.2
5.8	24.0		9.0	24.2		12.2	16.0

Table SRXB-82-1 (continued)
Core Boric Acid Concentration vs. Time

Time (hours)	Core Concentration (wt%)		Time (hours)	Core Concentration (wt%)		Time (hours)	Core Concentration (wt%)
12.3	15.7		15.5	9.9		18.7	7.2
12.4	15.5		15.6	9.8		18.8	7.2
12.5	15.3		15.7	9.7		18.9	7.1
12.6	15.0		15.8	9.6		19.0	7.1
12.7	14.8		15.9	9.5		19.1	7.0
12.8	14.6		16.0	9.4		19.2	7.0
12.9	14.4		16.1	9.2		19.3	6.9
13.0	14.2		16.2	9.1		19.4	6.9
13.1	13.9		16.3	9.0		19.5	6.8
13.2	13.7		16.4	8.9		19.6	6.8
13.3	13.5		16.5	8.8		19.7	6.7
13.4	13.3		16.6	8.7		19.8	6.7
13.5	13.1		16.7	8.6		19.9	6.6
13.6	12.9		16.8	8.6		20.0	6.6
13.7	12.7		16.9	8.5		20.1	6.6
13.8	12.6		17.0	8.4			
13.9	12.4		17.1	8.3			
14.0	12.2		17.2	8.2			
14.1	12.0		17.3	8.1			
14.2	11.8		17.4	8.1			
14.3	11.7		17.5	8.0			
14.4	11.5		17.6	7.9			
14.5	11.4		17.7	7.8			
14.6	11.2		17.8	7.8			
14.7	11.0		17.9	7.7			
14.8	10.9		18.0	7.6			
14.9	10.7		18.1	7.6			
15.0	10.6		18.2	7.5			
15.1	10.5		18.3	7.4			
15.2	10.3		18.4	7.4			
15.3	10.2		18.5	7.3			
15.4	10.1		18.6	7.3			

SRXB-83 (RAI 2.8.5.6.3-6)

The use of Eq. 2.8.5.6.3-4 appears to underestimate the ANS 1971 decay heat standard. Staff calculations show the decay heat fraction at 3600 sec. to be 0.017735 compared to the CENPD-254 value of 0.01761. Please show that the expression reproduces the 1971 standard.

Response

The ANS 1971 standard documents the decay heat as a curve, not an equation or table of values. Because there are no specific decay heat values documented in this standard, the CENPD-254-P-A, Post-LOCA Long Term Cooling Evaluation Model, June 1980 values are based on an equation used to fit this curve. These values are within the uncertainty related to reading the curve. As the NRC calculation shows a discrepancy in the third significant figure of the decay heat fraction, this is judged to be within the allowed uncertainty, and therefore, the CENPD-254 values are considered acceptable for use in calculating the decay heat fraction.

To demonstrate the acceptability of the specific use of EPU LAR Attachment 5, Equation 2.8.5.6.3-4, the calculation was redone using the NRC calculated value of 0.017735 for 3600 seconds. The original calculation, using the CENPD-254 value, is documented in EPU LAR Attachment 5, Section 2.8.5.6.3.5.2. Equation 2.8.5.6.3-4 is used in the reactor vessel hydrostatic pressure balance calculation to calculate the core-to-break steam flow pressure drop. The differences in values can be seen in Table SRXB-83-1 below:

Table SRXB-83-1
Differences in Decay Heat Calculations

	CENPD-254	NRC Calculation
Decay heat fraction at 3600 seconds	0.01761	0.017735
Core boil off rate (lbm/sec)	52.15	52.52
Core-to-break steam flow pressure drop (psi)	0.628	0.637
Hydrostatic head of the outlet plenum (psi)	4.982	4.973

For the mixing volume, only 3.308 psi is credited for the hydrostatic head of the outlet plenum. Thus, the difference in hydrostatic head between the CENPD-254 methodology and NRC calculation values is well within the available margin in the static head of outlet plenum. Therefore, there are no changes to the results of the analysis, and the calculations done using CENPD-254 decay heat fraction are judged to be acceptable.

<u>SRXB-84 (RAI 2.8.5.6.3-7)</u>

The BAM tanks are assumed to discharge following the limiting large break LOCA with respect to boric acid precipitation? Do the tanks fully discharge or is there EOP guidance is given regarding termination of the BAM tanks following a LOCA? Please explain. What is the minimum lower plenum fluid temperature prior to recirculation and is this temperature below the precipitation limit? What is the earliest and latest liming for the switch to recirculation? Please explain.

<u>Response</u>

The boric acid precipitation analysis conservatively assumes complete discharge of the boric acid makeup (BAM) tanks. This is a conservative departure from the emergency operating procedure (EOP) guidance, which realigns the charging pump suction from the BAM tanks to an alternate source when the BAM tanks are at a level of between 20-30%.

BORON analysis results show that, using a conservatively large initial volume for the BAM tanks, they empty at approximately 2.3 hours. The code conservatively assumes saturation temperature at 14.7 psia, which minimizes the precipitation limit. The assumed lower plenum temperature prior to recirculation is 212°F. The switch to recirculation occurs after the refueling water tank (RWT) and BAM tanks empty. The BAM tank empties after the RWT, so the switch to recirculation will occur at approximately 2.3 hours post-LOCA.

<u>SRXB-85 (RAI 2.8.5.6.3-8)</u>

What is the uncertainty in flow rates for the flow split between the hot and cold leg injection and was this taken into account? At 6 hrs, the hot side injection is equal to the boil-off rate. If the hot side injection is less than boil-off, flushing will not begin until sufficient flow in excess of the boil-off is injected if the switch is made at 4 hrs. If the switch is made at 4 hrs, the cold side does not appear to be able to match boil-off based on the results of Fig. 2.8.5.6.3-73. Core uncover could then result. Please explain.

<u>Response</u>

The high pressure safety injection (HPSI) pumps provide simultaneous hot and cold side injection. The boric acid precipitation analysis results assume a conservative minimum required hot side injection flow rate of 250 gpm and a cold side injection flow rate of 273 gpm. Flow delivery analysis has been done to confirm that, including uncertainty, with the HPSI system aligned for simultaneous hot and cold side injection, a minimum flow of 273 gpm can be provided for cold leg injection and a minimum flow of 275 gpm can be provided for hot leg injection. The analysis does not define a specific flow split uncertainty. However, the flow splits are explicitly calculated in a conservative manner, modeling the actual piping configurations and accounting for the uncertainties. For a hot leg break, 273 gpm of cold leg injection is available to replenish the liquid in the vessel. For a cold leg break, 454.75 gpm (100% of the 250 gpm hot side injection and 75% of the 273 gpm cold side injection) is available to replenish the liquid in the vessel. Therefore, even with taking flow split uncertainty into account, the minimum required injection flow to preclude core uncovery and boric acid precipitation can be delivered as discussed below.

For cold leg breaks, with simultaneous hot and cold side injection initiated at 6 hours, hot side injection (250 gpm) approximately equals the boil-off rate at 6 hours, and therefore, flushing begins post 6 hours. The analysis credits flushing only after hot side injection exceeds the boil-off rate. Flushing is equal to the hot side injection minus the boil-off rate. Vessel inventory is not depleted for cold leg breaks with simultaneous hot and cold side injection starting at 4 hours since safety injection flow to the vessel consists of hot leg injection plus three quarters of the cold leg injection. Thus, for cold leg breaks, boric acid precipitation is precluded and the core remains covered.

EPU LAR Attachment 5, Figure 2.8.5.6.3-73 shows core boil-off versus time. As seen in this figure, core boil-off is equal to 273 gpm at approximately 4.3 hours. For hot leg breaks, as described above, there is 273 gpm of cold side injection available to replenish the core boil-off flow. When simultaneous hot and cold side injection is begun at 4 hours for hot leg breaks, there is some vessel inventory depletion between 4 to 4.3 hours on the order of 6 ft^3. This decrease in inventory from 4 to 4.3 hours is small and has a negligible impact on the results. Beyond 4.3 hours, cold side injection exceeds the boil-off rate, thus flushing flow equal to cold side injection minus the boil-off rate is assured. Therefore, for hot leg breaks, boric acid precipitation is also precluded and the core remains covered.

SRXB-86 (RAI 2.8.5.6.3-9)

What is the effect of axial power shape on precipitation timing? Bottom peaks reduce the liquid inventory in the mixing volume. Please provide the most bottom skewed axial power distribution, and justify the axial shape used in the analysis.

Response

In the boric acid precipitation analysis, the axial power shape is only used in calculating the mixing volume for boric acid precipitation. The justification in the analysis for use of the flat power shape is as follows:

> A flat axial power shape is selected as a reasonably conservative representation of the axial power distribution. In the boric acid precipitation analysis, a bottom peaked shape (i.e., positive axial shape index (ASI)) is in a conservative direction since it results in more bubbles being produced lower in the core and, consequently, more level swell (i.e., higher void fraction). In general, long term axial power shapes start out at beginning of cycle as cosine shapes and transition to saddle shapes later in cycle. These shapes are generally fairly symmetrical and, hence, have ASIs that are close to 0. A flat axial power shape is a conservative representation of this fact since it maximizes the power at the bottom of the core.

To evaluate the impact of bottom peaked shapes, using a conservative bottom peaked axial power shape with ASI equal to +0.2, the mixing volume calculation was recalculated. Using this power shape resulted in approximately a 2% reduction in the mixing volume. This will have a negligible effect on the results of the boric acid precipitation analysis. The bottom peaked axial power shape used for this calculation is documented in Table SRXB-86-1 below:

Table SRXB-86-1

Mixing Volume Region*	Height (feet)	Axial Peaking Factor
1	9.715	0
2	1.1392	1.2
3	1.1392	1.3
4	1.1392	1.3
5	1.1392	1.2
6	1.1392	1.1
7	1.1392	1.0
8	1.1392	0.9
9	1.1392	0.8
10	1.1392	0.7
11	1.1392	0.5
12	7.962	0

Note

* Mixing Volume Region 1 is the lower plenum, and Region 12 is the outlet plenum.

SRXB-87 (RAI 2.8.5.6.3-9)

If the RCS refills and disperses the boric acid for breaks less than 0.036 ft^2 shown in Fig. 2.8.5.6.3-75, please explain the last column that shows breaks 0.012 ft^2 and smaller uncovering.

Response

The CELDA analysis shows that breaks 0.012 ft^2 and smaller need to be cooled down with the shutdown cooling (SDC) system and cannot be cooled down with simultaneous hot and cold side injection. That is, these breaks are sufficiently small and do not have the capacity by themselves to remove the decay heat energy from the system. The system will pressurize and require SDC for long term decay heat removal.

There are two basic procedures in the long-term cooling plan, namely simultaneous hot and cold side injection (large break procedure) and SDC (small break procedure). For the long term cooling analysis, using the CELDA code, if the break is small enough for the reactor coolant system (RCS) to refill with safety injection water, then SDC is applicable. In SDC, the RCS is cooled down via the steam generators to the SDC entry temperature and SDC is initiated. Decay heat is then removed by the SDC system. The largest small break for SDC is 0.036 ft^2 according to the CELDA analysis.

The CELDA analysis also shows that breaks 0.013 ft^2 and larger can be cooled down with simultaneous hot and cold side injection. Simultaneous hot and cold side injection is applicable when the break flow is sufficiently large to remove the decay heat from the RCS in the long term with simultaneous hot and cold side injection. The high pressure safety injection (HPSI) pump replenishes the RCS inventory that is lost out the break. Steam generator heat transfer may need to be maintained for a period of time until decay heat drops sufficiently and the energy flow out the break is sufficient to remove decay heat. Then the system will reach an equilibrium pressure in which the break flow rate equals the simultaneous hot and cold side injection flow rate. In this case, the flow out the break is capable of removing the total energy added to the system by decay heat.

SRXB-88 (RAI 2.8.5.6.3-11)

What provisions in the long term cooling plant are made in the event shutdown cooling is unavailable. Please explain.

Response

The break spectrum for long term cooling is analyzed to assure that for each break size, at least one of two success paths are achievable – either the plant can enter shutdown cooling (SDC) or simultaneous hot and cold side injection can adequately cooldown the plant. This analysis was performed for a two atmospheric dump valve (ADV) cooldown. The decision to enter SDC must be made prior to 16 hours post-loss of coolant accident (LOCA).

For a two ADV cooldown, breaks 0.013 ft^2 and larger can be cooled through simultaneous hot and cold side injection, and breaks 0.036 ft^2 and smaller can enter SDC. St. Lucie Unit 2 has two trains available for SDC. The worst single failure assumes the loss of an emergency diesel generator (EDG); thus, even with this assumed failure, one train will always remain available for SDC, so the smallest breaks can be cooled.

A related question about SDC being available (SRXB-97) was discussed at the FPL/NRC public meeting on the St. Lucie EPU LARs held on September 21, 2011. Based on that discussion, it was requested that FPL confirm that the residual heat removal (RHR) system has two trains and is available for initiation following a small break LOCA assuming a limiting single failure. The following response was provided, and is appropriate here as well.

Under 10 CFR 50, Appendix K methodology, the worst single failure must be assumed for any analysis. For the post-LOCA long term cooling analysis, the worst single failure assumed is the failure of an EDG, which results in the loss of one train (high pressure safety injection (HPSI) and low pressure safety injection (LPSI) pump).

Thus, for the loss of an EDG, one LPSI pump is still available to support SDC. Therefore, SDC will remain available, even with the failure of an EDG.

The analysis methodology shows that all break sizes can be adequately cooled down in the event of a LOCA by demonstrating that:

1) the break is small enough such that the reactor coolant system (RCS) can be refilled and the SDC entry temperature can be met prior to depletion of the condensate storage tank (CST) inventory, and thus the SDC system is used to remove decay heat and prevent boric acid precipitation (small breaks), or

2) the break is large enough for the break flow and simultaneous hot and cold side injection with a HPSI removes decay heat and prevents boric acid precipitation in the long-term (large breaks).

The results show that there is an overlap of break sizes such that both methods are acceptable (i.e., both SDC and maintaining hot and cold side injection).

The analysis assumes that one emergency core cooling system (ECCS) train is available. Note that St. Lucie Unit 2 has two ECCS trains. Accordingly, it is assumed that SDC is available and one HPSI for simultaneous hot and cold leg injection is available. Thus, there is no need to enter into a feed and bleed mode of cooldown.

SRXB-89 (RAI 2.8.5.6.3-12)

Please demonstrate that the larger small breaks, up to 1.0 ft2 are not limiting. Also, what are the results of a severed injection break with degraded injection into the intact lines since the flow to the break through the broken cold leg will exceed 25% of the total injected flow. Please provide the results of the broken injection line break.

Response

The EPU small break loss of coolant accident (SBLOCA) analysis spectrum includes the 0.06 ft^2/pump discharge (PD) break size which demonstrates that for this size break and larger, the injection pressure of the safety injection tanks (SITs) will be reached. Therefore, the EPU SBLOCA analysis demonstrates for small break sizes of 0.06 ft^2/PD and larger, the SITs will inject into the reactor coolant system (RCS) and in conjunction with the injection from high pressure safety injection (HPSI)/low pressure safety injection (LPSI) pumps and a charging pump, will recover the core and terminate the heatup of the cladding before the cladding temperature approaches the peak cladding temperature (PCT) of the limiting 0.05 ft^2/PD break. The safety injection (SI) pump flow is from a single train since the worst single failure for the SBLOCA event is a failure of an emergency diesel generator.

The most limiting small break location has been historically demonstrated to be at the bottom of the reactor coolant pump (RCP) discharge leg which the EPU SBLOCA analysis has analyzed. The RCP discharge leg is the limiting break location because it maximizes the amount of spillage from the emergency core cooling systems (ECCS). All analyzed break sizes model SI flow on the affected broken loop to spill to containment, including that from the SIT injection. A severed injection inlet nozzle line break is 0.5592 ft^2 which is significantly larger than the 0.06 ft^2 and would have significant SIT injection and SI flow including LPSI, since the RCS pressure would be expected to achieve a level below that of the LPSI shutoff head. Due to the size of the injection line break, and considering that SIT injection flow and SI pump flow spilled would be higher than 25% total flow for a severed injection line, this break would still have significant SI flow and SIT injection and as discussed above, would have cladding temperatures that are bounded by the limiting small break. It is also noted that the St. Lucie Unit 1 analysis of a severed injection line break showed non-limiting results. St. Lucie Units 1 and 2 have the same size injection line (~10 in diameter) and similar HPSI/LPSI performance characteristics. Unit 2 has a higher SIT pressure than Unit 1 which will result in earlier SIT flow. Table SRXB-89-1 provides a comparison of pertinent St. Lucie Units 1 and 2 parameters.

Therefore, no explicit analysis of a severed injection line is performed.

Table SRXB-89-1
St. Lucie Units 1 and 2 Comparison

	Unit 1	Unit 2
Low Pressure Reactor Trip (psia)	1807	1810
SI Actuation System (SIAS) (psia)	1520	1646
Minimum SIT pressure (psia)	244.7	499.7
Minimum HPSI flow	Table SRXB-89-2	Table SRXB-89-3

Table SRXB-89-2
St. Lucie Unit 1
HPSI Minimum Flowrate (4 Loop)

RCS Pressure (psia)	Flowrate (gpm)
15	616.3
315	531.3
615	424.4
815	339.8
1015	219.9
1115	119.5
1125	102.0
1135	66.4
1145	6.7
1145.5	0.0

Table SRXB-89-3
St. Lucie Unit 2
HPSI Minimum Flowrate (4 Loop)

RCS Pressure (psia)	Flowrate (gpm)
0	608.4
217	551.6
393	500.4
551	448.8
699	393.2
829	340.4
943	290.4
1035	244.4
1104	194.4
1177	109.6
1198	76.8
1205	63.6
1212	0

<u>**SRXB-90 (RAI 2.8.5.6.3-13)**</u>

What is the cause of the temperature spike just prior to PCT for the clad temperature plots. What causes the temperature to then drop very rapidly. Please explain.

<u>**Response**</u>

The small break loss of coolant accident (SBLOCA) analyses for Combustion Engineering (CE) design pressurized water reactors (PWRs) using the NRC accepted SBLOCA CENPD-137 evaluation model is conservatively performed at the burnup with the maximum initial fuel rod stored energy. This maximizes the fuel centerline temperatures for the transient.

To account for all rod internal pressures (RIP) predicted to occur during the lifetime of the fuel, a RIP study is performed to determine a RIP that calculates rupture to occur at the time of peak cladding temperature (PCT). The additional energy deposition due to the zirc-water reaction of the additional cladding surface exposed due to rupture causes a spike in cladding temperature and maximizes the PCT. The sharp drop in the spike is caused by steam cooling. The overall drop in cladding temperature is caused by core recovery due to safety injection.

SRXB-91 (RAI 2.8.5.6.3-14)

Please explain why the limiting small break PCT decreased 40 F at the EPU higher power conditions.

Response

EPU LAR Attachment 5, Table 2.8.5.6.3-10 documents a decrease in peak cladding temperature (PCT) of 40°F for the limiting break size of 0.05 ft^2 in the EPU analysis relative to the analysis of record (AOR).

The EPU analysis has a number of changes relative to the current AOR, most notably, the implementation of the replacement steam generators (RSGs), the implementation of the EPU, a revised and more limiting high pressure safety injection (HPSI) delivery curve, and the crediting of charging flow. A reduction in PCT was expected with the implementation of the RSGs because of an increase in heat transfer area and fluid volume. RSGs were not credited in the AOR. An increase in PCT was expected with the implementation of the EPU because of an increase in power and the more limiting HPSI delivery curve. Also, the crediting of charging flow provides a benefit for the PCT results. Additional benefit is also obtained due to the reduction in the steam generator tube plugging level from 30% (AOR) to 10% (EPU analysis). Thus, due to the overall impact of these competing changes, the PCT decreased by 40°F for the limiting break.

SRXB-92 (RAI 2.8.5.6.3-15)

Please explain why the PCTs decreased for the large breaks analyzed at EPU conditions.

Response

The following differences in plant configuration and large break loss of coolant accident (LBLOCA) initial conditions between the EPU and the analysis of record (AOR) LBLOCA emergency core cooling system (ECCS) performance analyses were reviewed to explain the observed reduction in the peak cladding temperature (PCT) for EPU.

Core Power. The core power for the EPU analysis is 3030 MWt and for the AOR is 2754 MWt. All things being equal, core power tends to increase the PCT. The higher core power for the EPU analysis compared to the AOR analysis represents a PCT penalty on the order of 54°F.

Limiting Safety Injection Tank (SIT) Configuration. The limiting SIT configuration for the EPU analysis is minimum pressure, maximum liquid inventory, minimum temperature, and maximum flow resistance. The limiting SIT configuration for the AOR is maximum pressure, maximum liquid inventory, minimum temperature, and maximum flow resistance. The maximum and minimum values for these SIT parameters are unchanged between the AOR and EPU analyses, but the particular configuration of these parameters that is associated with the limiting PCT has changed. There are no general conclusions regarding which SIT configuration tends to consistently produce higher PCTs because of the synergistic effects of the many other initial and boundary conditions that are simultaneously analyzed.

Physics Parameters. Two core physics parameters which strongly affect the calculated PCT are the maximum integrated radial peaking factor ($F_{r,max}$) and a radiation enclosure (for the hot rod) parameter (*X*-factor).

$F_{r,max}$ is a core physics parameter which defines the power of the hot assembly relative to the core average assembly. *X*-factor is a radiation enclosure parameter representing the power distribution among fuel rods in the rod-to-rod thermal radiation model used in the 1999 evaluation model (EM) for radiation heat transfer calculations on the hot rod. The *X*-factor is a measure of the flatness in power of the radiation enclosure surrounding the hot rod. A larger *X*-factor indicates a larger variation in power between the hot rod and its surroundings which benefits radiation heat transfer. Conversely, a lower *X*-factor indicates a lower variation in power between the hot rod and its surroundings which minimizes radiation heat transfer.

The maximum integrated radial peaking factor for the EPU analysis is 1.6 and for the AOR analysis is 1.7. The lower maximum integrated radial peaking factor for the EPU analysis compared to the AOR analysis represents a PCT benefit on the order of 53°F.

The *X*-factor for the EPU analysis is 1.5 and for the AOR analysis is 1.8. The more conservative representation of the radiation enclosure in the EPU compared to the AOR analyses represents a PCT penalty on the order of 19°F.

Containment Spray (CS) Flow Rates. The CS flow rate affects the calculated containment pressure during the refill phase of a LBLOCA. Higher CS flow rates produce lower containment pressures which decrease reflood rates due to increased steam venting resistance. The total CS flow rate for the EPU analysis is 9,000 gpm and for the AOR analysis is 6,900 gpm. The increased CS flow rate for the EPU analysis compared to the AOR analysis represents a PCT penalty on the order of 23°F.

ECCS Flow Rates. ECCS flow rates affect the calculated containment pressure during a LBLOCA. The worst single failure scenario for a LBLOCA using the CE 1999 EM is no failure of an ECCS component due to the maximum spillage of ECCS water into containment that is calculated to occur. The total ECCS flow rate for the EPU analysis is 10,942.4 gpm and for the AOR analysis is 7,480 gpm. The higher ECCS flow rate for the EPU compared to the AOR analyses represents a PCT penalty on the order of 9°F.

Initial Reactor Coolant System (RCS) Flow Rates. The RCS mass flow rate for the EPU analysis is 40,072 lbm/sec and for the AOR analysis is 35,796 lbm/sec. The higher initial RCS mass flow rates for the EPU compared to the AOR analyses represents a PCT benefit on the order of 17°F.

Discretionary Conservatism. Discretionary conservatism is defined as conservatism that is added to an analysis to provide additional margin. It is not required by the analysis methodology. Consequently, it may be decreased or removed in future revisions of the analysis or in future work that builds on the analysis. It is also available to trade-off against the impact that future changes may have on the results of the analysis.

Discretionary conservatism is typically incorporated through (a) conservative decreases in the third reflood rate resulting in lower reflood heat transfer coefficients, (b) conservative decreases in the core two-phase level during the less than 1 in/sec reflood period resulting in less steam generation in the two-phase mixture region, and thus less steam flow used for the calculation of the steam cooling heat transfer coefficients, and (c) additional surface area to the containment passive heat sink representation.

A comparison of the discretionary conservatisms applied to the EPU and AOR analyses showed that the AOR analysis has greater conservatisms applied to the third reflood rate and the core two-phase level during the less than 1 in/sec reflood period, and the EPU analysis has a slightly greater total passive heat sink surface area.

The effects of discretionary conservatisms (a) and (b) are to increase the calculated PCT by reducing the reflood rate and depressing the core two-phase mixture level, respectively, which reduces the calculated steam cooling heat transfer coefficients for the hot rod. The effect of discretionary conservatism (c) is to increase PCT by lowering containment pressure which increases steam venting resistance which, in turn, reduces the reflood rate. However, the effects of (a) and (b) are significantly greater than (c) on PCT.

The amount of discretionary conservatisms (a) and (b) applied in the AOR analysis produced an increase in PCT for the limiting case on the order of 80°F. In contrast, the amount of discretionary conservatisms (a) and (b) applied in the EPU analysis only produced an increase in PCT for the limiting case on the order of 20°F.

With all discretionary conservatisms applied, the EPU PCT for the limiting case is less than the AOR PCT by 17°F. When the EPU and AOR analyses are both stripped of discretionary conservatism (a) and (b) the EPU PCT for the limiting case is *greater* than the AOR PCT by 63°F.

The discretionary conservatism applied to the EPU compared to the AOR analyses represents a ΔPCT benefit on the order of 60 °F.

Summary. Table SRXB-92-1 below summarizes the EPU PCT impact for the limiting case. Key parameters in the AOR and EPU analyses were identified and the impact that each difference in the parameter has on PCT was calculated separately using the 1999 EM. Some of the PCT impacts are a penalty (resulting in an increase in PCT for EPU) while others are a benefit (resulting in a decrease in PCT for EPU), but the overall impact on PCT due to the parameter differences is a calculated benefit of 22°F. This calculated benefit agrees well with the reported benefit of 17°F in the EPU analysis. The difference is due to the synergistic effects these parameter differences have on PCT when combined in a single case and the impact of other less significant parameters (*e.g.*, RCS wall heat).

Table SRXB-92-1
Summary of EPU PCT Impact

Plant Configuration or Boundary/Initial Condition Parameter	Change From AOR	Effect on PCT Due to Changes From AOR	Limiting Case Change In PCT Due to Changes From AOR[a] (°F)
Penalties			
Core power	Increase	Increase (Penalty ↑)	+54
Rod-to-rod thermal radiation hot rod enclosure	Decrease	Increase (Penalty ↑)	+19
Containment spray flow rates	Increase	Increase (Penalty ↑)	+23
ECCS flow rates	Increase	Increase (Penalty ↑)	+9
Containment passive heat sink areas	Increase	Increase (Penalty ↑)	+3
Benefits			
Discretionary conservatism	Decrease	Decrease (Benefit ↓)	-60
Maximum integrated radial peaking factor	Decrease	Decrease (Benefit ↓)	-53
Initial RCS flow rate	Increase	Decrease (Benefit ↓)	-17
Summation of penalties			+108
Summation of benefits			–130
Overall simulated impact[b]		Decrease (Benefit ↓)	-22
Overall actual impact[c]		Decrease (Benefit ↓)	-17

NOTES

(a) The change in PCT is calculated as EPU PCT – AOR PCT for each parameter in the table. A positive number indicates that the calculated PCT for EPU is greater than for AOR and is identified as a *penalty*. Conversely, a negative number indicates that the calculated PCT for EPU is less than for AOR and is identified as a *benefit*.

(b) The *simulated* impact is the sum of the individual parameter impacts as calculated using the CE 1999 EM.

(c) The *actual* impact is the difference between the limiting PCT reported in the AOR and EPU analyses and represents the effects of all the parameters simultaneously.

SRXB-93 (RAI 2.8.5.6.3-16)

If the containment is being vented during a LOCA, what is the impact on PCT? Please explain.

Response

The peak cladding temperatures (PCTs) calculated with the Combustion Engineering (CE) 1999 evaluation model (EM) (References SRXB-93-1 through SRXB-93-6) are sensitive to the core reflood rates that are calculated to occur during a large break loss of coolant accident (LBLOCA) simulation. The core reflood rate, in turn, is sensitive to the containment pressures that are calculated to occur during the simulation. In general, lower containment pressures have an adverse effect on the core reflood rate due to the increased specific volume of steam in the reactor coolant system (RCS) associated with lower containment pressures and the accompanying increased steam venting resistance. This adverse effect tends to increase PCTs.

Containment pressures during a LBLOCA first increase to a maximum value and then, decrease with time due to the steam condensation effects and air cooling of the containment spray (CS) system and the emergency core cooling system (ECCS) water spillage. Containment pressures peak during the blowdown phase, and decrease during the refill and reflood phase of a LBLOCA. Simulations with the CE 1999 EM of a double-ended guillotine break at the reactor coolant pump (RCP) discharge have shown that the case of no single failure of a safety grade system during the transient contribute to higher PCTs, precisely because of the maximum containment steam condensation and air cooling effects that the maximized CS system flow rates and ECCS water spillage produces. Therefore, any effects that reduce the containment pressures initially or during the LBLOCA are expected to contribute to higher PCTs.

Containment venting during a LBLOCA would cause the containment mass of gas (air and steam) to be reduced with time. The reduced mass of air and steam in containment would reduce containment pressure and, therefore, contribute to increased PCTs for the same reasons that the mechanisms of steam condensation, air cooling, and increased steam venting resistance have on PCT as described above.

The COMPERC-II/LB computer program (References SRXB-93-7 through SRXB-93-9) calculates the containment pressure during a LBLOCA in the CE 1999 EM. This current application of the code does not have an explicit model to simulate the effects of a changing containment mass and pressure due to venting during the simulation. Currently, COMPERC-II/LB can only conservatively account for the effects of containment venting by the specification of a reduced initial containment pressure. The reduction in containment pressure due to the containment purge/venting prior to the isolation of the purge system at approximately 5 seconds is evaluated to be insignificant and well within the margin available in the pressure uncertainty included in the initial containment pressure. The LBLOCA analysis thus covers the impact on pressure characterized by the purging of air from containment due to the containment purge system operating for about 5 seconds after the break.

The LBLOCA analysis also has conservatisms included in the maximum containment volume, containment heat sinks and maximum containment spray flow, which all result in conservatively low containment pressure during the transient.

Thus the impact of venting the containment for a finite period of time is adequately covered in the LBLOCA analysis and no separate PCT impact is determined.

References

SRXB-93-1 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

SRXB-93-2 CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

SRXB-93-3 CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

SRXB-93-4 CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

SRXB-93-5 CENPD-132, Supplement 4-P-A, "Calculative Methods For the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

SRXB-93-6 CENPD-132-P-A, Supplement 4-P-A, Addendum 1-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," August 2007.

SRXB-93-7 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

SRXB-93-8 CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

SRXB-93-9 CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

SRXB-94 (RAI 2.8.5.6.3-17)

Please explain, the basis for, and justify the maximum RWST temperature of 104 F used in the small break analysis.

Response

EPU LAR Attachment 5, Table 2.8.5.6.3-6 indicates that a maximum refueling water storage tank, refueling water tank (RWT) at St. Lucie, of 104°F is an input parameter for small break loss of coolant accident (SBLOCA) emergency core cooling system (ECCS) performance analysis.

Technical Specification (TS) 3/4.5.4 Refueling Water Tank LCO 3.5.4.c states that the borated water solution temperature for Modes 1 to 4 is to be between 55°F and 100°F. The uncertainty for the nominal maximum RWT temperature is ±4°F.

The RWT is the water source for the high pressure safety injection (HPSI)/low pressure safety injection (LPSI) and charging flows credited in the EPU SBLOCA calculations. A higher RWT water temperature is more conservative, relative to the cooling ability of the water delivered by the HPSI/LPSI and charging pumps and hence, the TS RWT maximum temperature plus uncertainty (100°F + 4°F = 104°F) is used in the SBLOCA calculations.

SRXB-95 (RAI 2.8.5.6.3-18)

Please provide the analysis results for the worst break with considerations to downcomer boiling. Describe the worst single failure and justify the downcomer nodalization used in the evaluation.

Response

1. Worst Break Size and Worst Single Failure

Case studies were performed to identify the worst case from the perspective of downcomer boiling for EPU conditions. Cases were run and results surveyed to identify the worst case from two perspectives – the case which produced the maximum downcomer temperature and the case which produced the minimum temperature margin to saturation. The results show that the worst case from either perspective is essentially the same and is characterized by the following conditions:

- 1.0 double ended guillotine break at the reactor coolant pump (RCP) discharge (DEG/PD);
- Failure of an emergency diesel generator (EDG);
- Minimum pressure in the safety injection tank (SIT);
- Maximum liquid volume in the SIT;
- Maximum temperature in the SIT;
- Minimum SIT discharge coefficient; and
- Maximum refueling water tank (RWT) liquid temperature.

In contrast, the EPU peak cladding temperature (PCT) limiting case for ZIRLO® clad fuel is characterized by the following conditions:

- 0.6 double ended guillotine break at the RCP discharge (DEG/PD);
- No failure of the emergency core cooling system (ECCS);
- Minimum pressure in the SIT;
- Maximum liquid volume in the SIT;
- Minimum temperature in the SIT;
- Maximum SIT discharge coefficient; and
- Minimum RWT liquid temperature.

Table SRXB-95-1 shows the analysis results of the EPU downcomer boiling limiting cases and a comparison to the EPU PCT limiting case. The EPU downcomer boiling limiting cases were identified from two perspectives: the case which produced the maximum liquid temperature and the case which produced the minimum liquid subcooling margin in the downcomer. Table SRXB-95-1 shows the downcomer conditions corresponding to the time at which the maximum liquid temperature and the minimum liquid temperature margin to saturation were calculated to occur.

2. Justification of Downcomer Nodalization

The COMPERC-II/LB computer code (References SRXB-95-1 through SRXB-95-3) is used in the Combustion Engineering (CE) 1999 evaluation model (EM) (References SRXB-95-4 through SRXB-95-9) to perform the reactor coolant system (RCS) refill and reflood hydraulic analysis and to calculate the containment minimum pressure.

The COMPERC-II/LB nodalization consists of four major regions of the RCS. These are (1) the downcomer and lower plenum, (2) the core bypass, (3) the core region and upper plenum, and (4) the steam flow network external to the vessel. The downcomer and lower plenum region consists of the entire reactor vessel annulus between the vessel and the core barrel and also includes the lower plenum up to the bottom of the active fuel. The downcomer and lower plenum region contains completely mixed subcooled water. Heat addition to the coolant in the downcomer and lower plenum from the reactor vessel wall and internals is considered. Mass and energy balances are used to compute the quantity of coolant in the lower plenum and the enthalpy and density of that coolant as a function of time.

The issue of downcomer boiling has not been explicitly addressed in the past for CE plants. However, studies were performed in the past to examine downcomer boiling nodalization and cold leg condensation effects in the COMPERC-II/LB computer code. These studies identified the following key results which justify the downcomer nodalization used in the COMPERC-II/LB for 10 CFR 50 Appendix K type large break loss of coolant accident (LBLOCA) ECCS performance analyses in view of downcomer boiling concerns.

(1) Downcomer boiling was concluded not to be a concern for CE plants partly because (a) all CE plants have dry atmosphere containments, (b) all CE plants have large SITs (accumulators) with long delivery times (e.g., 30 to 60 seconds), and (c) CE plants have high capacity safety injection (SI) pumps. These plant features reduce the susceptibility to downcomer boiling.

(2) An evaluation for two typical CE designed plants, and for St. Lucie Unit 2, with the current EM showed that the downcomer is significantly subcooled during and at the end of a typical LBLOCA calculation (See Table SRXB-95-1).

(3) The EM assumes full condensation during the discharge of the SITs. This overestimates the downcomer temperature during the time of SIT discharge. As noted above, the SITs for CE plants are sufficiently large to fill the downcomer with a minimum initial liquid SIT volume.

(4) The COMPERC-II/LB computer code wall heat model is a very conservative model using a semi-infinite slab presentation at a constant uniform initial temperature (vessel wall) which calculates wall heat to the downcomer fluid using a constant film coefficient and heat transfer area.

(5) No degradation due to two phase downcomer boiling effects during the reflood calculations were calculated for any of the St. Lucie Unit 2 EPU cases analyzed; that is, no deduction of the downcomer head due to two-phase effects with the consequent reduction of reflood rates and increases in PCT were calculated.

(6) The licensed methodology for CE designed plants for reflood thermal-hydraulics, in particular, the COMPERC-II/LB computer code, contains 10 CFR 50 Appendix K type assumptions designed to calculate conservatively low core reflood rates leading to conservatively high PCTs. In addition, the methods for prescribing the design

inputs related to ECCS equipment and worst case single failure requirements are also very conservative. The downcomer representation in COMPERC-II/LB as one large volume node prohibits the calculation of thermal stratification which could allow the upper regions of the downcomer to approach saturation temperatures earlier in the transient than would occur if the entire volume had to be heated to that temperature. Likewise, subcooled boiling is also simplified because the bubbles that can be generated along hot surfaces, even though the bulk water temperature is below saturation temperature, are not modeled. Neglecting these voids during reflood can over-estimate the gravitational head in the downcomer. However, as noted above, the COMPERC-II/LB 10 CFR 50 Appendix K methodology is very conservative for reflood thermal-hydraulics and PCT, and is adequate to compensate for the effects associated with downcomer boiling phenomena that are not explicitly represented in COMPERC-II/LB.

References

SRXB-95-1 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

SRXB-95-2 CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

SRXB-95-3 CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

SRXB-95-4 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

SRXB-95-5 CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

SRXB-95-6 CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

SRXB-95-7 CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W designed NSSS," June 1985.

SRXB-95-8 CENPD-132, Supplement 4-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," March 2001.

SRXB-95-9 CENPD-132-P-A, Supplement 4-P-A, Addendum 1-P-A "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 2007.

Table SRXB-95-1
Comparison of EPU Limiting Peak Cladding Temperature (PCT) Case and EPU Downcomer Limiting Cases

	Limiting PCT Case		Downcomer Boiling Limiting Cases	
	Data at Time of Maximum Temperature	**Data at Time of Minimum Temperature Delta to Saturation**	**Data at Time of Maximum Temperature**	**Data at Time of Minimum Temperature Delta to Saturation**
Peak cladding temperature (°F)	2087.3		1908.1	1977.2
Peak local oxidation (PLO) (%)	14.39		6.83	8.68
Maximum core wide oxidation (%)	0.954		0.496	0.654
Pressure (psia)	20.5	19.9	23.8	23.7
Temperature (°F)	131.2	130.3	177.1	177.0
Saturation temperature (°F)	228.9	227.5	236.6	236.5
Temperature margin to saturation (°F)	97.7	97.1	59.5	59.5

<u>SRXB-96 (RAI 2.8.5.6.3-19)</u>

What is the impact EPU conditions on the RCP trip criteria and timing evaluated for the limiting SBLOCAs? Please provide the results of the analysis of RCP trip for small breaks and demonstrate that the EOPs have been updated to reflect RCP trip at EPU conditions.

<u>Response</u>

The response is being provided in a separate submittal.

<u>SRXB-97 (RAI 2.8.5.6.3-20)</u>

How does failure to isolate an SIT impact the timing for establishing RHR and can RHR be successfully initiated for all small breaks that refill prior to exhaustion of the CST given this failure? Please provide an analysis to demonstrate RHR can be initiated prior to CST exhaustion for the limiting small break.

<u>Response</u>

This question was discussed at the FPL/NRC public meeting on the St. Lucie EPUs held on September 21, 2011. Based on that discussion, the question was clarified to request FPL to confirm that the residual heat removal (RHR) system has two trains and is available for initiation following a small break loss of coolant accident (LOCA) assuming a limiting single failure. The following response is provided.

Under 10 CFR 50 Appendix K methodology, the worst single failure must be assumed for any analysis. For the post-LOCA long term cooling analysis, the worst single failure assumed is the failure of an emergency diesel generator (EDG), which results in the loss of one train (one high pressure safety injection (HPSI) pump and one low pressure safety injection (LPSI) pump). The second train of one HPSI pump and one LPSI pump is not affected and is available.

Thus, for the loss of an EDG, one LPSI pump is still available to support shutdown cooling (SDC). It was concluded at the meeting that the fact that SDC will remain available, even with the failure of an EDG, provides sufficient response to this RAI.

SRXB-98 (RAI 2.8.5.6.3-21)

For small break in the approximately range 0.002 to 0.005 ft2, the RCS will refill early trapping hot fluid in the pressurizer. Subsequent to the RCS cooldown, the operators will throttle HPSI flow to initiate RHR once entry pressure and temperature conditions are achieved. With hot water trapped in the pressurizer, what procedure is used to reduce RCS pressure and initiate shutdown cooling before the CST is exhausted? What qualified equipment, if any, is used to reduce RCS pressure? Please show an analysis that demonstrates RHR/shutdown cooling can be successfully initiated for very small breaks to establish a long term cooling mode of heat removal.

Response

The response is provided in two parts: a procedures and qualified equipment evaluation, followed by an analysis evaluation.

i. Procedures and Qualified Equipment

EOP-3 Loss of Coolant Accident (LOCA) provides the actions for responding to a LOCA including steps to reduce reactor coolant system (RCS) pressure and initiate shutdown cooling (SDC) before the condensate storage tank (CST) is exhausted. EOP-3 is applicable to the scenarios postulated in the RAI, including LOCAs with hot water trapped in the pressurizer.

Section 4.0 Operator Actions Step 18 provides the steps to cooldown the RCS to SDC. Step 19 provides the steps to depressurize the RCS to SDC. Step 30 provides the steps for RCS void elimination.

Step 18 provides direction to cooldown the RCS to SDC using the steam bypass control system.

- If reactor coolant pumps (RCPs) are operating, use the General Operating Procedure (GOP) - Reactor Plant Cooldown - Hot Standby to Cold Shutdown.
- If RCPs are not operating, use the Abnormal Operating Procedure – Natural Circulation Cooldown.
- Contingency actions are provided for cooling down using the atmospheric dump valves or the auxiliary feedwater system.

Step 19 provides direction for depressurizing the RCS to SDC.

- Depressurize the RCS using main pressurizer spray (RCPs operating) or auxiliary pressurizer spray (charging pumps).
- If safety injection (SI) throttling criteria has been satisfied, control pressure by throttling SI flow.

Step 30 provides direction for removing voids from the RCS, if required.

- If the RCS fails to depressurize and voiding is suspected, minimize or secure letdown and stop depressurizing the RCS. Then raise and lower RCS pressure using pressurizer heaters and spray and charging and SI pumps. Operate the head vents as necessary. Monitor pressurizer and reactor vessel level for inventory trends.
- If depressurization is not possible and voiding is suspected in the steam generator (SG) tubes, steam the suspected SG using the steam bypass control system or atmospheric dump valves (ADVs), feed and bleed the SG using feedwater and SG blowdown system, monitor pressurizer and reactor vessel level for inventory trends.

All systems and components required are qualified. These steps ensure that SDC can be initiated prior to exhausting the CST.

ii. <u>Analysis Evaluation</u>

For the post-LOCA long term cooling analysis, SDC entry temperature is 300°F and entry pressure is 275 psia.

All of the equipment listed above in Item i. is available to reduce RCS pressure per the EOPs and confirmed to be safety grade.

Post-LOCA long term cooling is presented in EPU LAR Attachment 5, Section 2.8.5.6.3.6. The analysis is performed using NRC approved methodology (CENPD-254-P-A, Post-LOCA Long Term Cooling Evaluation Model, June 1980). The decay heat removal portion of the methodology uses the CEPAC, NATFLOW, and CELDA codes. According to the post-LOCA long term cooling methodology, the following steps are taken to ensure that shutdown cooling mode can be successfully entered for small breaks:

> For sufficiently small breaks, the RCS refills, the HPSI pumps maintain system pressure, and the RCS liquid level is sufficient for entry into the SDC mode. The RCS temperature is then checked to assure that SG cooling has reduced it to the SDC entry value. Next, the HPSI pumps are realigned to discharge entirely to the cold legs; they are then throttled to reduce the RCS pressure to the SDC entry value (after venting the safety injection tanks (SITs) if these have not yet discharged). The shift is then made to the SDC mode.

Results of the CELDA long term cooling analysis show that breaks smaller than 0.036 ft^2 can be cooled down with the SDC system. In addition, the CELDA analysis shows that the 0.005 ft^2 break refills in 5.1 hours and reached the SDC entry temperature of 300°F at approximately 6.7 hours post-LOCA. The 0.002 ft^2 break refills in 4.2 hours and reached the SDC entry temperature of 300°F at approximately 7.5 hours post-LOCA. For these small breaks, hot water may refill and be trapped in the pressurizer. Although the CELDA analysis does not explicitly model the refill of the pressurizer with hot water, the cooldown of the loops calculated by CELDA is still applicable. That is, breaks larger than 0.002 ft^2 can be cooled down to 300°F in less than 7.5 hours (the time to exhaust the CST inventory calculated by CEPAC is at least 17 hours). Although CELDA does not explicitly model the hot water in the pressurizer, the procedure in the EOPs Step 18 for natural circulation cooldown, described above, demonstrates that the cooldown and depressurization of the system after the pressurizer refills can be completed with the CST inventory. EPU LAR Attachment 5, Section 2.8.7.2 showed that there is sufficient inventory in the CST to complete a natural circulation cooldown (NCC) with the EPU. For the small breaks considered here (0.002 ft^2 to 0.005 ft^2), the break also removes some of the energy from the system during the cooldown. Thus, the CST inventory is sufficient to complete the NCC and depressurize the system with the small breaks considered above.