FPL Energy
Seabrook Station

FPL Energy Seabrook Station
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October 12, 2004

Docket No. 50-443
SBK-L-04072

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

References:

1. FPL Energy Seabrook, LLC letter NYN-04016, "LAR 04-03, Application for Stretch Power Uprate," dated March 17, 2004.
2. FPL Energy Seabrook, LLC letter NYN-04032, "Background Information to Support LAR 04-03, Application for Stretch Power Uprate," dated April 1, 2004.
3. FPL Energy Seabrook, LLC letter NYN-04016, "Response to Request for Additional Information Regarding LAR 04-03, Application for Stretch Power Uprate," dated May 26, 2004.
4. FPL Energy Seabrook, LLC letter SBK-L-04044, "Changes to LAR 04-03, Application for Stretch Power Uprate," dated September 13, 2004
5. NRC letter to FPL Energy Seabrook, LLC, "Seabrook Station Unit 1 – Request for Additional Information for Proposed Amendment Request Regarding the Application for Stretch Power Uprate (TAC MC2364)" dated August 18, 2004.
6. Revised Request for Additional Information RAI #22 provided by fax, dated August 23, 2004.
7. Four Additional Requests for Additional Information RAIs #96 through #99 provided by e-mail, September 1, 2004.

Seabrook Station
Response to Request for Additional Information Regarding
License Amendment Request 04-03, Application for Stretch Power Uprate

By letter dated March 17, 2004 (Reference 1), FPL Energy Seabrook, LLC (FPL Energy Seabrook) requested an amendment to facility operating license NPF-86 and the plant technical specifications for Seabrook Station. This license amendment request (LAR) is an application for a stretch power uprate which will increase the Seabrook Station licensed reactor core power by 5.2% from 3411 megawatts thermal (MWt) to 3587 MWt. This LAR is supported by additional information submitted to the NRC by References 2, 3, and 4.

A001

In References 5, 6, and 7, the NRC has requested additional information to support its review of Seabrook Station LAR 04-03. The enclosures to this letter contain FPL Energy Seabrook's responses to the requests for additional information (RAIs) provided in your correspondence.

Westinghouse Electric Company has identified proprietary information in the responses to RAIs 2, 24, 25, and 36. The non-proprietary responses to the RAIs are contained in Enclosure 1 to this letter. The proprietary information for the responses to RAIs 2, 24, 25, and 36 is contained in Enclosure 2. The application for withholding proprietary information from public disclosure including an affidavit in conformance with the provisions of 10 CFR 2.790 for withholding proprietary information is contained in Enclosure 3.

Commitments made by FPL Energy Seabrook are provided in Enclosure 4.

Should you have any questions concerning this information, please contact Mr. Stephen T. Hale, Power Uprate Project Manager, at (603) 773-7561.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

cc. S. J. Collins, NRC Region I Administrator
S. P. Wall, NRC Project Manager, Project Directorate I-2
G. T. Dentel, NRC Resident Inspector

Mr. Bruce Cheney, Director
New Hampshire Bureau of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301

OATH AND AFFIRMATION

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC hereby affirm that the information and statements contained in the responses to the request for additional information to support the review of License Amendment Request 04-03 are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
Before me this

12 day of October, 2004

Michael O'Keefe
Notary Public

Mark E. Warner
Site Vice President

MICHAEL D. O'KEEFE
MY COMMISSION EXPIRES FEB. 5, 2008
NOTARY PUBLIC
NEW HAMPSHIRE

Enclosure 1 to Letter No. SBK-L-04072
Response To Request for Additional Information
for LAR 04-03, Application for Stretch Power Uprate
(Non-Proprietary)

This enclosure contains:

- Responses to RAIs 1 through 99
- Attachment RAI 2-1
- Attachment RAI 4-1
- Attachment RAI 42-1

**REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT REQUEST 04-03
APPLICATION FOR STRETCH POWER UPRATE
SEABROOK STATION
DOCKET NO. 50-443**

Instrumentation and Controls

RAI #1

Please confirm that the following are the only safety-related instrumentation setpoint changes required for the SPU:

a. Technical Specification (TS) Table 2.1-1, "Reactor Trip System Instrumentation Trip Setpoints," Functional Unit 13, "Steam Generator Water Level Low-Low," is revised as follows:
   1) The Trip Setpoint is changed from >14.0% to >20.0% of narrow range instrument span.
   2) The Allowable Value is changed from >12.6% to >19.5% of narrow range instrument span.
   3) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

b. TS Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," Functional Unit 5b, "Turbine Trip, Steam Generator Water Level High-High (P14)," is revised as follows:
   1) The Trip Setpoint is changed from >86.8% to >90.8% of narrow range instrument span.
   2) The Allowable Value is changed from >87.7% to >91.3% of narrow range instrument span.
   3) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

c. TS Table 3.3-4, Functional Unit 6a, "Feedwater Isolation, Steam Generator Water Level Hi-Hi (P14)," is revised as follows:
   1) The Trip Setpoint is changed from >86.8% to >90.8% of narrow range instrument span.
   2) The Allowable Value is changed from >87.7% to >91.3% of narrow range instrument span.
   3) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

d. TS Table 3.3-4, Functional Unit 7c, "Emergency Feedwater, Steam Generator Water Level Low-Low, Start Motor-Driven Pump and Start Turbine-Driven Pump," is revised as follows:

1) The Trip Setpoint is changed from >14.0% to >20.0% of narrow range instrument span.
2) The Allowable Value is changed from >12.6% to >19.5% of narrow range instrument span.
3) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

FPL Energy Seabrook Response:

FPL Energy Seabrook letter to NRC SBK-L-04044 dated September 13, 2004 identified and corrected errors in the License Amendment Request concerning the "Steam Generator Water Level High-High (P14)" Trip Setpoint and Allowable Value in items b and c above. The Trip Setpoint was revised from "≥86.8% to ≥90.8%" to "≤86.0% to ≤90.8%" and the Allowable Value was revised from "≥87.7% to ≥91.3%" to "≤87.7% to ≤91.3%".

In addition to the listed setpoints in items a, b, c, and d above, the Overtemperature ΔT and Overpower ΔT trip setpoints are changing for the SPU as noted in LAR Attachment 1, Subsection 4.3.3.2, "Margin to Trip Analysis" and Table 4.3-2 (pages 4-26 and 4-31, respectively). The changes for the Overtemperature ΔT and overpower ΔT trips setpoints are as follows:

	Current Value	SPU Value
Overtemperature ΔT	K1 ≤ 1.18	K1 ≤ 1.21
Overpower ΔT	K4 ≤ 1.121	K4 ≤ 1.116

RAI #2

In each case where a setpoint is shown as a percentage of instrument span, identify the instrument by manufacturer, model and range, its span, and the actual and allowable physical values of the setpoint.

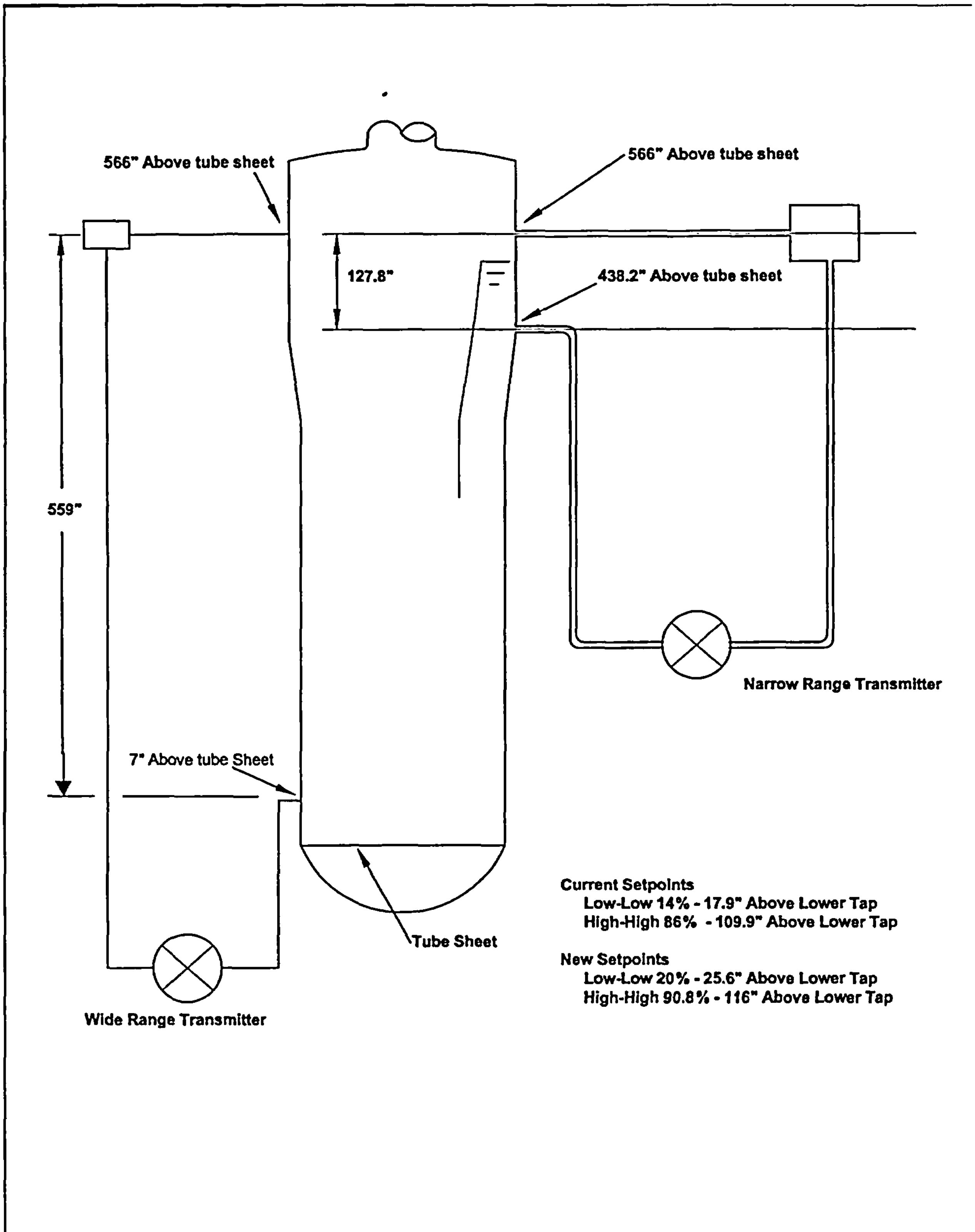
FPL Energy Seabrook Response:

For steam generator level, the transmitters are Rosemount model 1154DP4RA and the process racks are Westinghouse 7300 racks. The narrow range is defined as the distance between the lower instrument tap and the upper instrument tap. The distance is 127.8 inches – see Figure RAI 2-1. The transmitter range is 150 inches of water column (W.C.) and the calibrated span is 85.72 inches of water column. The process rack range and span is 0 – 10 volts D.C. The physical unit of measure for this function is level. Based on the tap to tap distance of 127.8 inches, the previous setpoint of 14% narrow range span is 17.89 inches above the lower tap. The setpoint change for the SPU is 20% of narrow range span which is 25.6 inches above the lower tap.

For Overtemperature ΔT and Overpower ΔT function, there are multiple inputs. The Reactor Coolant System resistance-temperature detectors (RTDs) are Weed model N9004E-2B, the pressurizer pressure transmitters are Rosemount model 1154GP9RA, the Nuclear Instrumentation System input is Westinghouse nuclear instrumentation system neutron detectors and process racks, and the protection racks are Westinghouse 7300 racks. The physical unit of measure for this trip is ΔT. The span of the ΔT trips is ΔT equivalent to 150 percent power. The setpoint is a variable setpoint and therefore does not have fixed ΔT span value. The Allowable Value is expressed in percent of ΔT span. Additional information is provided in the tables in Attachment RAI 2-1 (non-proprietary) and Attachment RAI 2-2 (proprietary) relative to the span for each input to the trip.

Westinghouse proprietary information is provided in Enclosure 2.

Figure RAI 2-1



RAI #3

Provide calculations and supporting setpoint methodology for the setpoints indicated in Attachment 2 of the March 17, 2004, submittal. Details should be sufficient to allow the Nuclear Regulatory Commission (NRC) staff to understand the values used, assumptions made, and formulae used. If the NRC staff has previously reviewed and approved the setpoint methodology, provide a reference to the acceptance document.

FPL Energy Seabrook Response:

Attachments RAI 2-1 (non-proprietary) and RAI 2-2 (proprietary) contain tables which provide the calculation results for the safety related setpoints that are changing for the SPU. Westinghouse proprietary information is provided in Enclosure 2.

RAI #4

Explain why, in each case, the Total Allowance (TA), Z, and Sensor Error (S) setpoints indicated in TS Table 2.1-1 and Table 3.3-4 were changed from values to N.A.

FPL Energy Seabrook Response:

Based on the operability determination for the setpoints changing for the SPU, the values marked as N.A. are no longer applicable. The Technical Specifications are being changed from the five-column method to the two-column method. In the five-column method, these values were calculated to help in the evaluation of the operability of the channel in the event that the Allowable Value was exceeded. The setpoints changing for the SPU follow the two column method, thus, the Nominal Trip Setpoint and Allowable Value will be the only values in the plant technical specification used for channel operability determination. If the setpoint being tested exceeds the Allowable Value, it will be declared inoperable and appropriate action will be taken. Therefore, the remaining columns are no longer needed and are marked as N.A. A further explanation of operability determination is provided in Attachment RAI 4-1. It should be noted that the Allowable Values determined for the SPU were not calculated using Method 3 of ISA S67.04.02.

RAI #5

Explain how channel operability is determined for the functional units indicated in Question 1 above.

FPL Energy Seabrook Response:

The operability determination process and application to the plant technical specifications is provided in Attachment RAI 4-1.

RAI #6

Due to the SPU, are there any other changes to the instrumentation and controls needed beyond the setpoint changes identified in Question 1 above (i.e., changes in control systems, span changes, or instrument replacement)?

FPL Energy Seabrook Response:

LAR Attachment 1, Section 10.6, "Modifications" (page 10-7), provides a listing of instrumentation and controls changes required for the SPU.

Subsequent to submittal of the LAR, the designs for the modifications for the SPU were completed which resulted in the following changes to the list:

Additions:

- Change low suction pressure trip setpoint for feedpumps.
- Change the time delay setpoint for the reactor coolant pump undervoltage relays.

Deletions:

- Re-span steam flow transmitters was determined not to be necessary.
- Modify scaling and indicator for generator megawatt meter was determined not to be necessary.

RAI #7

Discuss any changes to how Seabrook meets the acceptance criteria and guidelines outlined in NRUEG-0800, Standard Review Plan (SRP), Chapter 7, "Instrument and Controls," because of the SPU.

FPL Energy Seabrook Response:

There were no changes made on how Seabrook Station meets the acceptance criteria and guidelines outlined in NUREG-0800, Standard Review Plan, Chapter 7, "Instrumentation and Controls," due to the SPU. Also, as noted in LAR Attachment 1, Subsection 4.3.1, "Reactor Protection System / Engineered Safety Features Actuation System," (page 4-21), the applicable design criteria are not affected by the SPU.

Electrical Engineering

RAI #8

In support of Section 8.4.16.7, "Grid Stability," provide details about the grid stability analysis including assumptions and results and conclusions for the power uprated condition.

FPL Energy Seabrook Response:

The Independent System Operator (ISO) for New England requires all additions to the power grid to undergo a detailed system impact study. A copy of the "Power Systems Energy Consulting, Seabrook Uprate System Impact Study, Phase 1 Final Report," dated January 22, 2004 was transmitted to NRC in FPL Energy Seabrook letter NYN-04032 dated April 1, 2004.

The System Impact Study evaluated the effect of the uprate addition on MVAR support, pre-contingency and post-contingency voltage criteria, and stability performance. The report concluded that the uprate would have no significant adverse impact on thermal or voltage performance for the system conditions and contingencies that were studied. In addition, the stability of the system will remain adequate using currently established administrative protocols for generator output management.

Based on the conclusions of the System Impact Study for this uprate, all Seabrook Station and NEPOOL voltage and stability requirements will be met without any hardware replacements in the plant or the grid.

<u>RAI #9</u>

Address and discuss the following points:

a. Identify the nature and quantity of MVAR support necessary to maintain post-trip loads and minimum voltage levels.

b. Identify what MVAR contributions Seabrook is credited for providing to the offsite power system or grid.

c. After the power uprate, identify any changes in MVAR quantities associated with Items a. and b. above.

d. Discuss any compensatory measures to adjust for any shortfalls in Item c. above.

e. Evaluate the impact of any MVAR shortfall listed in Item d. above on the ability of the offsite power system to maintain minimum post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges with the transmission system operator.

<u>FPL Energy Seabrook Response:</u>

See FPL Energy Seabrook response to RAI #8.

Additionally, specific responses to RAI questions 9a, b., c., d., and e. are as follows:

a. The current MVAR support necessary to maintain post-trip house loads and a minimum voltage of 345 kV is 28 MVAR.

b. Seabrook Station generator is capable of supplying 560 MVAR at the current gross electrical output of approximately 1206 MWe. Note that this MVAR capability is not a grid requirement, but simply the maximum capability of the generator.

c. After the SPU, the MVAR support necessary to maintain post-trip house loads and a minimum voltage of 345 kV is 29 MVAR. The post SPU MVAR capability requirement set by ISO-NE is 367 MVAR.

d. In accordance with the System Impact Study, no compensatory measures, such as capacitor banks, will be necessary for the reduced generator MVAR capability. Standard and current administrative protocols will remain in place to ensure optimum grid stability.

e. As stated in Item d. above, ISO-NE has not identified a shortfall in the MVAR production or the capability to maintain the requisite minimum voltage of 345 kV.

Vessel and Internal Integrity and Welding

RAI #10

In Section 5.1.3.5, "Pressurized Thermal Shock," of the March 17, 2004, submittal, FPL Energy Seabrook, LLC (FPLE or the licensee) states the following:

> Based on this evaluation, the reference temperature-pressurized thermal shock values will remain below the Nuclear Regulatory Commission screening criteria values using the projected SPU fluence values through end of license [EOL] for 40 Effective Full Power Years [EFPYs] for Seabrook Station and thus meet the requirements of 10 CFR 50.61.

To substantiate the statement, provide the following:

a. The projected neutron fluence (E > 1.0 Mev) for each vessel beltline material at EOL including the impact of the proposed SPU.
b. Reactor vessel beltline material properties including initial RT_{NDT}, Cu and Ni contents and the source of the information (generic or plant specific).
c. RT_{PTS} values for the all vessel beltline materials. Also provide the basis of RT_{PTS} values.

FPL Energy Seabrook Response:

a. The projected 40 Effective Full Power Years (EFPY) fluence used for all beltline materials is 2.20×10^{19} n/cm^2 (E > 1.0 MeV). The corresponding fluence factor is 1.21. The reference temperature – pressurized thermal shock (RT_{PTS}) values were calculated using the SPU fluence values. The 40 EFPY RT_{PTS} values for all beltline materials is well less than the pressurized thermal shock screening criteria, and thus are acceptable.

b. The data requested is provided in Table RAI 10-1 below:

TABLE RAI 10-1
REACTOR VESSEL BELTLINE MATERIAL PROPERTIES

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}[(b)] (°F)
Intermediate Shell Plate R-1806-1[(d)]	0.045	0.61	40
Intermediate Shell Plate R-1806-2[(d)]	0.06	0.64	0
Intermediate Shell Plate R-1806-3[(d)]	0.075	0.63	10
Lower Shell Plate R-1808-1[(d)]	0.06	0.58	40
Lower Shell Plate R-1808-2[(d)]	0.06	0.58	10
Lower Shell Plate R-1808-3[(d)]	0.07	0.59	40
Beltline Weld Seams (Heat # 4P6052)[(a)]	0.047	0.049	-60[(e)]
Seabrook Unit 1 Surveillance Weld (Heat # 4P6052)[(c)]	0.02	0.075	---

Notes:

(a) The beltline weld seams consist of the intermediate shell longitudinal welds (101-124A,B, C), the lower shell longitudinal welds (101-142A,B, C) and the intermediate to lower shell girth weld (101-171). These welds were fabricated with wire heat No. 4P6052, flux type 0091, flux lot No. 0145. The copper and nickel weight percents were taken from CE Reports NPSD-1039, Rev. 2 & NPSD-1119, Rev. 1. These Cu & Ni values do not match RVID2 however, they would produce a more conservative table chemistry factor versus those in RVID2. This fact is negligible since the welds are not limiting.

(b) The initial RT_{NDT} values are measured values unless otherwise noted.

(c) Average of the two data points presented in Table A-3 of WCAP-10110.

(d) Average of Lukens Mill Test Report and CE Test (Documented in WCAP-10110).

(e) Measured value documented in WCAP-10110

c. The basis for the RT_{PTS} values (as shown below) are ΔRT_{PTS} and Initial RT_{NDT}. The ΔRT_{PTS} values are calculated using chemistry factor (CF) and fluence factor. The chemistry, which is used to develop the CF, is shown in the previous response along with the initial RT_{NDT}. The lower shell plate (R-1808-3) and the beltline welds also used credible surveillance data with CF of 39.5°F and 12.4°F, respectively. The methodology basis for RT_{PTS} was 10 CFR 50.61.

TABLE RAI 10-2
REACTOR VESSEL BELTLINE MATERIAL RT_{PTS} VALUES

Material Description	ΔRT_{PTS} (°F)	Margin (°F)	RT_{PTS} (°F)
Intermediate Shell Plate R-1806-1	34.5	34	109
Intermediate Shell Plate R-1806-2	44.8	34	79
Intermediate Shell Plate R-1806-3	57.5	34	102
Lower Shell Plate R-1808-1	44.8	34	119
Lower Shell Plate R-1808-2	44.8	34	89
Lower Shell Plate R-1808-3	53.2 47.8	34 17[a]	127 105
Beltline Weld Seams (Heat # 4P6052)	37.1 15.0	37.1 15.0[a]	14 -30

Notes:

(a) Credible surveillance data

RAI #11

In Section 5.1.3.5, "Results," under "Upper Shelf Energy," FPLE states the following:

> Based on this evaluation, the upper shelf energy [USE] values for Seabrook Station will maintain a level above 50 ft-lbs.

For each beltline material, provide the USE values at the end of the current licensed life, including the impact of the SPU. Also, provide the basis of the calculation including beltline material copper percentage, the unirradiated USE value, and the projected neutron fluence (E>1.0 MeV) at 1/4 thickness. If surveillance data was used, provide the surveillance data.

FPL Energy Seabrook Response:

The upper shelf energy (USE) values were calculated using the SPU fluence values. The 40 EFPY USE using the SPU fluence projections are acceptable. See Table RAI 11-1 for the 40 EFPY USE using SPU fluence projections:

TABLE RAI 11-1
REACTOR VESSEL BELTLINE MATERIAL RT_{PTS} VALUES

Material Description	40 EFPY 1/4T Fluence	Unirradiated USE	Projected Decrease	40 EFPY USE
Intermediate Shell Plate R-1806-1	1.31	82	20	66
Intermediate Shell Plate R-1806-2	1.31	102	20	82
Intermediate Shell Plate R-1806-3	1.31	115	20	92
Lower Shell Plate R-1808-1	1.31	78	20	62
Lower Shell Plate R-1808-2	1.31	77	20	62
Lower Shell Plate R-1808-3	1.31	78	20	62
Beltline Weld Seams (Heat # 4P6052)	1.31	156	20	125

RAI #12

In Section 5.1.3.5, under "Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves," FPLE states the following:

> This review indicates that the revised adjusted reference temperature [ART] the SPU will be less restrictive than that used in developing the current adjusted reference temperature values for Seabrook Station at 20 [EFPY].

To substantiate the statement, provide the following:

a. Basis for current Pressure-Temperature (P-T) limits (applicability in EFPY and ART values at the 1/4 thickness (T) and 3/4T locations).

b. Projected ART values for the proposed period of applicability using the SPU fluence.

FPL Energy Seabrook Response:

a. The basis for the current pressure-temperature (P-T) curves is WCAP-15745. These curves used a fluence of 1.324 x 10^{19} n/cm^2 (E > 1.0 MeV) for 20 Effective Full Power Years (EFPY). The limiting material was the lower shell plate R-1808-1 with a 1/4T and 3/4T adjusted reference temperature (ART) of 109°F and 88°F, respectively. The SPU projected fluence for 20 EFPY was 1.12 x10^{19} n/cm^2 (E > 1.0 MeV), thus the current pressure-temperature curves from WCAP-15745 are still valid and conservative under the SPU for a period up to 20 EFPY.

b. No adjusted reference temperature values were calculated for the SPU since the fluence projection was less for the SPU than those used in WCAP-15745 for the development of the Seabrook Station pressure temperature limit curves in question. See WCAP-15745 for the adjusted reference temperature values.

<u>RAI #13</u>

Table Matrix-1 of NRC Review Standard RS-001, Revision 0, provides the NRC staff's basis for evaluating the potential impacts for stretch power uprates and the subsequent aging effects. In Table Matrix-1, the staff states that, in addition to the SRP, guidance on the neutron irradiation-related threshold levels inducing irradiation-assisted stress corrosion cracking (IASCC) in reactor vessel (RV) internal components are given in Westinghouse document, License Renewal Evaluation, Aging Management for Reactor Internals, WCAP-14577, Revision 1-A. WCAP-14577,Revision 1-A establishes, a threshold of 1 x 10^{21} n/cm^2 (E $\geq$0.1 MeV) for the initiation of IASCC, loss of fracture toughness, and/or void swelling in pressurized water reactor (PWR) RV internal components made from stainless steel (including cast austenitic stainless steels) or Alloy 600/82/182 materials. In Table Matrix-1 of NRC Report RS-001, the staff established guidance that plants exceeding this threshold of neutron irradiation would either have to establish plant-specific degradation management programs for managing the aging effects associated with their RV internals or else indicate that the licensees would participate in industry programs designed for investigating and managing age-related degradation in the RV internal components. Provide the threshold fluence values for the internals (E > 0.1 MeV) due to the SPU. Also, discuss the inspection program that will be implemented by Seabrook if the threshold values exceed 1 x 10^{21} n/cm^2 (E $\geq$0.1 MeV).

<u>FPL Energy Seabrook Response:</u>

There are a number of industry activities currently underway to characterize and address aging effects on reactor vessel internals under the EPRI Materials Reliability Project (MRP). As a result of these efforts, further understanding of these aging effects will be developed by the industry over time that will provide additional bases for whether inspections over and beyond those currently required by ASME Section XI should be implemented. Additionally, FPL has been an active participant in the Westinghouse Owners Group (WOG) reactor vessel internals materials programs from their inception, particularly in the area of baffle/former bolting, including the Joint Owners Baffle Bolt (JOBB) program.

The MRP strategy is to evaluate potential aging mechanisms and their effects on specific reactor vessel internals parts by evaluating causal parameters such as fluence, material properties, state of stress, etc. Critical locations can thereby be identified and tailored inspections can be conducted on either an integrated industry, NSSS, or plant-specific basis.

The MRP projects include:

- Material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.
- Evaluation of the effects of irradiation, which include IASCC, swelling, and stress relaxation in pressurized water reactors.
- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on reactor vessel internals.
- Development of a long-term reactor vessel internals aging management strategy.

FPL has access to MRP products related to the reactor vessel as they are completed. Various tasks are addressed as JOBB program activities, which include a body of work to be performed by Electricite'de France. Note that FPL has committed to incorporate the results of these programs into the reactor vessel internals inspections as part of the renewed licenses for Turkey Point Units 3 and 4 and St. Lucie Units 1 and 2.

FPL Energy Seabrook commits to evaluate the results of the following EPRI MRP programs and to factor them into reactor vessel internals inspections as appropriate

- Material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.
- Evaluation of the effects of irradiation, which include IASCC, swelling, and stress relaxation in pressurized water reactors.
- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on reactor vessel internals.
- Development of a long-term reactor vessel internals aging management strategy

Piping Integrity and Nondestructive Examination

RAI #14

Discuss service adequacy of materials in nuclear steam supply system (NSSS) and balance-of-plant (BOP) piping under the power uprate operating conditions.

FPL Energy Seabrook Response:

As discussed in LAR Attachment 1, Subsection 5.11.3 (page 5-73), the change in the service conditions for the proposed SPU will not affect the service adequacy and/or performance of the nuclear steam supply system piping materials.

For balance of plant systems, the adequacy of the piping material (mostly carbon steel and chrome-moly) was evaluated for SPU operating conditions of pressure, temperature, fluid velocity, steam quality, chemistry, and where applicable, flashing conditions. The results concluded that there are no changes in system water chemistry as a result of SPU, and that the existing system pipe material, pipe size, and pipe wall thickness were appropriate and adequate for SPU conditions. SPU operating system pressures, temperatures and flows are bounded by existing system design. Refer to LAR 04-03, Attachment 1, Section 8.4 (page 8-13) "System Assessments" for individual system evaluations, Section 8.5.1 (page 8-92) "BOP Piping and Support" for piping and structural analysis evaluations, and Section 9.1.3 (page 9-2) "Flow Accelerated Corrosion Program" for impact of the SPU within the Flow Accelerated Corrosion Program. Additionally, clarification of the Flow Accelerated Corrosion Program was provided in FPL Energy Seabrook letter NYN-04047 dated May 26, 2004 (pages 35 and 36).

RAI #15

Discuss service adequacy of materials in control rod drive mechanism (CRDM) nozzles under the uprated conditions relative to primary water stress corrosion cracking susceptibility.

FPL Energy Seabrook Response:

As discussed in LAR Attachment 1, Subsection 5.11.3 "Materials Assessment" (page 5-73), there is no appreciable change in the primary water stress corrosion cracking susceptibility of the control rod drive mechanism nozzles from the SPU. Additionally, as noted in this LAR subsection, for Seabrook Station, the control rod drive mechanism nozzles are maintained at reactor vessel inlet conditions. Since the proposed SPU will result in a decrease in the vessel inlet temperature, the control rod drive mechanism nozzles will see less limiting conditions than currently.

Steam Generator (SG) Integrity & Chemical Engineering

RAI #16

In Section 5.7.4.4.3 of the March 17, 2004, submittal, when discussing tube undercut, the licensee states that the Seabrook tube end evaluation utilized the results from a previous evaluation of Model F SGs, adjusted, as appropriate, for Seabrook conditions.

Clarify the conservatism associated with the adjustment value being based on the increase in differential pressure across the tubesheet for the analyzed SPU conditions. In addition, confirm that all design criteria of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) are met for the tube and weld in the undercut condition.

FPL Energy Seabrook Response:

The adjustment value should be considered conservative for the calculation of the fatigue usage factor. The calculation of the fatigue usage factor is based on various transient combination stress range sets. Each set consists of two transients. A scaling factor was calculated for each transient based on the primary-to-secondary pressure differential loading increase. The transient combination stress range was scaled by the larger scaling factor that was associated with either of the two transients. This approach leads to a conservative fatigue usage factor.

The ASME Section III stress design criteria are met for the tube and tube weld in the tube undercut condition. The inside diameter of the tube may be machined to a reduction of 0.008 inches of wall thickness. There is no reduction in the tube weld.

RAI #17

In Section 5.7.4.4.5, the licensee indicates tube remnants in SG "D" from a 2002 tube removal were analyzed under the most limiting SPU conditions and determined to be stable with respect to fluid elastic excitation. The licensee states turbulence induced stresses are sufficiently low that crack propagation will not occur. The licensee concludes, as a result, that these tubes will not require stabilizers prior to operation at SPU conditions since the tubes have been shown to remain intact and will not contact any adjacent active tubes.

Clarify if the analysis indicates the tube remnants are acceptable for the remainder of the operating license without the need for stabilization or if stabilization will be performed at a later date but is not necessary prior to initial operation at SPU conditions.

FPL Energy Seabrook Response:

The analysis done for the SPU confirmed the tube remnants are acceptable for the remainder of the operating license without the need for stabilization.

RAI #18

In Section 5.7.4.4.6, when discussing the loose part wear evaluation in SG "B" and SG "C," the licensee states that the loose part will be evaluated on a cycle-by-cycle basis for future operation if tube wear is present.

Confirm that all loose parts will be evaluated on a cycle-by-cycle basis even if tube wear is not present. The NRC staff notes that a loose part's location can change during operation and that a part that has not caused wear in a certain location may potentially cause wear in a different location within the SG.

FPL Energy Seabrook Response:

Whenever the secondary side hand holes are opened for steam generator cleaning operations, a search for foreign objects is conducted. Parts that are detrimental to the tubes are removed, but in cases where the object cannot be removed, an analysis is performed which allows operation of the steam generators. In the case of the part in Steam Generator B, it was removed in the ninth refueling outage. In the case of the part in Steam Generator C, it has been lodged between two tubes since the first observance of the object in the first refueling outage. Each time the secondary hand holes are opened for cleaning operations, the part is confirmed in its original location. The tubes that have captured the object have been plugged since the first refueling outage. Analysis of potential loose parts detected by eddy current is performed on an inspection-to-inspection frequency.

RAI #19

Table 5.7.6-1 provides a summary of tube structural limits as determined by analysis for both high T_{avg} and low T_{avg} operating conditions. This analysis was performed assuming a uniform thinning mode of degradation in both the axial and circumferential directions.

For the locations and axial wear scar lengths shown in Table 5.7.6.1, confirm the following:

a. Uniform circumferential thinning, for analysis purpose, was assumed to be affecting 360 degrees of the tube;

b. The analytical approach for all locations shown in the table was identical but different axial lengths were assumed for the 360 degree thinning; and

c. For all cases involving normal and transient operating conditions (including postulated accidents with the appropriate safety factor on membrane and membrane plus bending loads) the most limiting structural limit resulted from maintaining a factor of safety of three against burst under normal steady state full power operation at the SPU conditions.

If any of items (a) through (c) above require clarification, confirm that the 40-percent plugging limit continues to provide adequate margin to the structural limit at the SPU conditions. That is, confirm the 40-percent plugging limit provides for a 360 degree, infinitely long flaw at the most limiting location (e.g., straight span, U-bend) under the most limiting condition (normal operating pressure, accident) with the appropriate regulatory margin on load (e.g. 3, 1.4, 1.2).

FPL Energy Seabrook Response:

a. Yes, the calculated structural limits for all locations are based on a uniform thinning mode of degradation affecting 360 degrees of the tube. The difference in the calculated structural limits is due to the variation in length of the degradation zones.

b. Yes, see the response to item a. above.

c. For the high T_{avg} condition, the most limiting structural limit resulted from both the normal operating condition loading and the postulated operating condition. That is, the resulting minimum wall requirement was the same for both the normal accident and postulated accident condition loading after all loads, including membrane and membrane plus bending loads, had been included with the appropriate safety factors, including the safety factor of 3 on burst. For the low T_{avg} condition, the most limiting structural limit resulted from the normal operating condition loading.

LAR Attachment 1, Table 5.7.6-1 (page 5-59) provides a summary of tube structural limits as determined by analysis for both high T_{avg} and low T_{avg} operating conditions. Structural limits are those conditions of a tube that will withstand the limiting loading conditions assuming ASME Code minimum material properties. The structural limit should not be confused with a plugging limit (i.e., 40% throughwall) which takes into account all relevant measurement and relational uncertainties. In practice, the 40% throughwall-plugging limit applies only for volumetric degradation, for example wear, which can be adequately sized by the qualified inspection techniques. Tubes with corrosion degradation (cracks) are plugged on detection, except where alternate repair criteria are licensed, because the sizing capabilities are generally inadequate to compare the degradation to a depth-based criterion. Both the condition monitoring assessment and operational assessment prepared at each inspection include documented allowances for uncertainties in the as-found and projected conditions of the tube to compare the structural limits defined in Table 5.7.6-1.

RAI #20

In Section 5.7.7.4.1, material considerations related to SG tube integrity are discussed. The licensee states that if the plant is operated at or near the analyzed maximum reactor vessel outlet temperature, the slight increase of maximum temperatures in the SG suggests a slight increase in the propensity for development of stress corrosion cracking (SCC) in comparison to the current parameters.

Did the evaluation of tube integrity at slightly higher temperatures consider tubes that may be more susceptible to SCC as a result of higher residual stress (see NRC Information Notice 2002-21, Supplement 1). The NRC staff understands all tubes exhibiting an eddy current offset have been removed from service. Assuming some tubes with higher residual stress may not exhibit an eddy current offset but be more susceptible to SCC, confirm that the existing SG tube inspection program accounts for changes in operating conditions and operating experience when determining the appropriate inspection intervals necessary to maintain tube integrity. Given the conclusion in Section 5.7.7.5 that the performance criteria of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 1, will continue to be met, discuss your plans for revising the Seabrook TSs to be consistent with the SG program and NEI 97-06.

FPL Energy Seabrook Response:

The NRC contention that high stress tubes may not be identified even after consideration of the manufacturing process and evaluation of the eddy current data screening data is not credible. For low row tubes, the last process prior to bending the U-bends is a thermal treatment process which, if performed, results in a very low stress state of the material. After bending, a final stress relief process is performed which renders the stress state in the U-bends to a very low value. If the tube was cold-worked after the pre-bend thermal treatment, but before the post bend stress relief was performed, the characteristic offset signal observed at Seabrook Station is created. To attain a high stress condition in both the straight leg and the U-bend in tubes less than row 11, both cold working after thermal treatment and failure to perform stress relief in the U-bends must occur. These are concurrent significant anomalies that would be prevented by the multiple manufacturing process controls. For example, failure to stress relieve a U-bend would result in many tubes with the same condition, since the stress relief process is performed on many tubes at the same time.

For long row tubes, it is the absence of a U-bend offset that indicates a potentially highly stressed tube. The last operation prior to bending is a thermal treatment of the entire length of the tube, which results in a very low stress state in the material. The bending process to form the U-bend strains the material in the U-bend region and causes the eddy current offset signal between the straight legs and the U-bend. If the tube was cold worked after the thermal treatment and before the bending process, the resulting offset will be very small or non-existent. The screening process identifies these tubes based on very conservative criteria. After U-bending, no other manufacturing processes that have the potential to cause elevated stresses in the entire length of the straight legs are applied. Therefore, it is considered non-credible that tubes with significantly elevated stress levels, outside the normal distribution of tubing conditions, could exist after the screening has been performed. Since the tubes remaining in service are manufactured according to the same procedures as all Alloy 600 TT (thermally treated) tubes, they are from the same population as all of the tubes.

Lead Model F plant (Wolf Creek) and F-type lead plant (Surry 2), both tubed with Inconel 600TT, continue to accrue service well beyond that of Seabrook Station without incidence of outer diameter stress corrosion cracking (ODSCC); the other U.S. plants using Alloy 600TT have also been free from outer diameter stress corrosion cracking, except for the Seabrook Station and Braidwood 2 observations. Excluding the tubes identified in Seabrook Station, Braidwood 2, and tubes identified later as other plants perform the screening for outer diameter stress corrosion cracking susceptibility, the in-service population of Alloy 600TT tubes in original or replacement steam generators constitute a set that encompasses the balance of the Seabrook Station tubes. The tubes included in this set are indistinguishable among themselves with regard to their residual stress levels.

The design T_{hot} values of several other Model F plants (620.6°F at Kori 3 and 4, Yonggwang 1 and 2) and 1 D5 Series steam generator (620.7°F at Catawba 2) are comparable to the SPU T_{hot} analyzed for Seabrook Station (621.4°F); a number of other domestic plants have design T_{hot} values in the 618°F to 619°F range. There have been no confirmed reports of cracking in these plants, though many of them have accrued more full power operation than Seabrook Station through 2003. Moreover the actual operating T_{hot} for Cycle 10 is 615°F, ~ 3° below the 618.2°F T_{hot} used for the reference case; thus there is little likelihood that the Cycle 11 post-SPU conditions will exceed those cited above for comparison. The calculated T_{hot} difference between the plants cited above and the analyzed Seabrook Station conditions is not significant with respect to observable differences in potential cracking.

All Alloy 600TT tubes in this set should be subject to the same rules with respect to the definition of inspection frequency intervals, periodic inspection samples, and inspection techniques. The changes attendant to the Seabrook Station SPU do not cause the Alloy 600TT tubes in Seabrook Station to operate under conditions not already experienced by other tubes in the industry population. Given that the susceptible tubes that can be identified have been removed from service, it is appropriate that Seabrook Station return to an inspection status for steam generators with Alloy 600TT not previously affected by cracking.

The operational assessment performed for Seabrook Station following the ninth refueling outage considers the effects of the proposed SPU on the degradation. Although the tubes that are identified as potentially susceptible to early outer diameter stress corrosion cracking due to an elevated stress condition have been plugged, the effects of the modified operating conditions appropriate to the SPU conditions were included in the cycle length analysis. The increased temperatures are reflected in the probability of detection and incidence of new indications, as well as in the applicable growth rate for the indications. For presumed corrosion mechanisms, the calculations were based on the Arrhenius equation with an activation constant of 35 Kcal/mole.

For all operators of steam generators tubed with Alloy 600TT and other advanced materials, the occurrence of outer diameter stress corrosion cracking is not, strictly speaking, unanticipated. Notwithstanding the excellent experience for the this material on an industry basis, Seabrook Station has regarded outer diameter stress corrosion cracking as a potential degradation mechanism; this understanding is documented in the Seabrook Station degradation assessments prepared prior to each outage. Qualified NDE techniques are demonstrated before each inspection, so that the likelihood of detecting initial outer diameter stress corrosion cracking indications is emphasized. Industry experience with unusual occurrence of tube degradation is rapidly disseminated and included in the degradation assessments to alert not just the NDE analysts but the service agencies and engineering personnel responsible for assuring continuing steam generator integrity.

FPL Energy Seabrook is monitoring industry activities associated with steam generators and will evaluate recommendations for implementation at Seabrook Station, including changes to the technical specifications, as appropriate.

RAI #21

The licensee concludes SG blowdown is acceptable for SPU conditions since the blowdown flow does not change and steam pressure and temperature are bounded by the Steam Generator System Design. Has the blowdown system experienced any degradation due to flow accelerated corrosion (FAC)? Did the blowdown system evaluation consider a potential increase in FAC due to a potential increase in particles (e.g., oxides) carried into the SG resulting from higher secondary system flow rates under SPU conditions?

FPL Energy Seabrook Response:

The Steam Generator Blowdown System is modeled in Seabrook Station's CHECWORKS program. A total of 30 locations are being inspected periodically on a multiple outage cycle. Piping in the system has not experienced any unusual degradation due to flow accelerated corrosion. Although there will no be changes in the Steam Generator Blowdown System operating parameters as a result of the SPU, the current CHECWORKS model will be reviewed and revised if required.

Input parameters to CHECWORKS do not specifically consider particles. However, any changes that result in increase wear due to SPU conditions would be identified by the periodic inspections discussed above.

RAI #22

In Section 9.1.3, the licensee provides a comparison of maximum FAC rates in the moisture seperator drain piping for current operating conditions and projected power uprate conditions. The licensee also indicates that, as part of the implementation of the SPU, the Electrical Power Research Institute (EPRI) CHECWORKS computer program will be revised to include the appropriate post-SPU operating parameters. Specifically, this revised model will be used to select inspection locations that will establish new baseline data for areas that may be susceptible to significant wear.

For current operating conditions, discuss any instances where CHECWORKS significantly under predicted actual wall loss due to FAC. Include any locations where CHECWORKS predicted no wall loss and significant wall loss occurred. Discuss any corrective actions taken as a result of FAC under prediction. If there were instances where FAC rates were significantly under predicted, discuss whether the SPU operating conditions could exacerbate this under prediction.

FPL Energy Seabrook Response:

There are no instances where CHECWORKS significantly under predicted wear. The Seabrook Station Flow Accelerated Corrosion Program is a relatively mature program that has significant flow accelerated corrosion inspection coverage. Historical inspection data has been incorporated in the CHECWORKS model to calibrate the predictive results through the use of a line correction factor. The line correction factor represents the degree to which the wear predicted by the CHECWORKS Pass 1 analysis is corrected to correlate with the wear determined from the actual ultrasonic testing (UT) data evaluation. A line correction factor of greater than one indicates that the Pass 1 model under-predicted wear, and the corrosion rates were increased to correlate with the measured wear. Inspection coverage was increased on those lines where CHECWORKS initially under-predicted wear to get a better understanding of wear in the line and to further refine and validate the analytical model.

The CHECWORKS model will be updated to reflect the SPU thermodynamic and flow conditions prior to restart from the spring 2005 refueling outage. A comparison of pre and post SPU predictions will be made to determine the impact of the SPU on flow accelerated corrosion wear rates. Additional inspection coverage will be considered for lines which indicate a significant change in predicted wear rates.

Civil and Engineering Mechanics

RAI #23

In several places, the application states that the allowable stresses after the SPU will exceed the ASME Code allowable stresses.

a. For the reactor vessel outlet nozzle safe end, the maximum stress exceeds $3S_m$. The footnote states that the stress intensities are qualified using simplified elastic-plastic analysis per Subsection NB of the ASME Code. Elaborate on the simplified elastic-plastic analysis.

b. For the RV bottom head instrument tubes, the maximum stress exceeds $3S_m$. There is no indication of how this exceedance was resolved (i.e., no footnotes). Elaborate on how this issue was resolved.

c. For the SG divider plate, the footnote to Table 5.7.2-1 states that plastic analysis performed in the design basis analysis for fatigue evaluation to show ASME Code criteria are satisfied. (Note that this is the condition both before and after SPU). Elaborate on the plastic analysis. What computer code was used for the analysis? Has this code previously been approved by the NRC?

d. For the SG tubesheet and shell junction, main feedwater nozzle, steam nozzle insert fillet weld, and support ring the stress exceeds $3S_m$. The footnotes state that simplified elastic plastic analysis was performed for the design basis evaluation, and K_e factors were used in the fatigue calculation to demonstrate that ASME Code criteria (NB-3228) are met. Elaborate on the elastic plastic analysis.

e. For the SG tube to tubesheet weld the stress exceeds $3S_m$. The footnote states that inelastic analysis performed in the original analysis demonstrated that ASME Code criteria are satisfied. Elaborate on the inelastic analysis.

FPL Energy Seabrook Response:

a. For the reactor vessel outlet nozzle safe end OD Location 2B, the $3S_m$ limit of 53.7 ksi for the SA-182, F316 austenitic stainless steel safe end material (at 508°F transient temperature) was exceeded by the calculated maximum range of stress intensity of 54.6 ksi.

   1. This maximum range of stress intensity resulted from the transient pair consisting of inadvertent depressurization (@ 500 seconds) and the shop hydrostatic test at 3107 psig. The range of stress intensity for this transient pair excluding thermal bending was calculated to be 44.37 ksi which is less than $3S_m = 53.7$ ksi as required for the simplified elastic-plastic analysis in NB-3228.3 of ASME Section III.

   2. The K_e factor for the stainless steel material was then calculated in accordance with NB-3228.3 using m = 1.7 and n = 0.3 as $3S_m < S_n < 3mS_m$ (that is: 53.7 ksi $< S_n =$ 54.6 ksi $<$ 91.29 ksi).

      So, $K_e = 1 + (1 - n)/[n(m - 1)](S_n/3S_m - 1) = 1.058$ ($K_e = 1.1$ was used in the fatigue analysis)

3. The procedure of NB-3227.6 (elastic analysis for stresses beyond the yield strength) did not need to be used.
4. The requirements for thermal ratcheting in NB-3222.5 were met. (Thermal stress range = 26.99 ksi < allowable range = 44.7 ksi)
5. The transient temperature for inadvertent depressurization of 508°F is < 800°F.
6. The ratio S_y/S_u = 0.28 is < 0.8

With K_e = 1.1 applied to the inadvertent depressurization/hydrostatic test transient pair in the fatigue analysis for Location 2B, the fatigue cumulative usage factor (CUF) was calculated to be 0.0. Alternatively, the maximum stress concentration factor of 5.0 was applied (even though there is no geometric discontinuity) to give a fatigue usage factor of 0.12 at Location 2B. The fatigue usage factor is less than 1.0.

b. For the bottom head instrumentation nozzle ID Location 1, the $3S_m$ limit of 69.9 ksi for the SB-166 Ni-Cr-Fe Alloy 600 material (throughout the design temperature range) was exceeded by the calculated maximum range of stress intensity of 70.67 ksi.

1. This maximum range of stress intensity resulted from the transient pair consisting of refueling (@ 2.0 hours) and a loss of power event. The range of stress intensity for this transient pair excluding thermal bending was calculated to be 64.38 ksi which is less than $3S_m$ = 69.9 ksi as required for the simplified elastic-plastic analysis in NB-3228.3 of ASME Section III.
2. The K_e factor for the Ni-Cr-Fe Alloy 600 material was then calculated in accordance with NB-3228.3 using m = 1.7 and n = 0.3 as $3S_m < S_n < 3mS_m$ (that is: 69.9 ksi < S_n = 70.67 ksi < 118.83 ksi).

   So, $K_e = 1+ (1 - n)/[n(m -1)](S_n/3S_m - 1) = 1.04$ (K_e = 1.04 was used in the fatigue analysis)

   The procedure of NB-3227.6 (elastic analysis for stresses beyond the yield strength) did not need to be used. Thermal ratcheting was not a concern.
3. The transient temperature 594°F does not exceed 800°F.
4. The ratio S_y/S_u = 0.44 is < 0.8

With Ke = 1.04 applied to the alternating stresses (S_a) in the fatigue analysis for Location 1, the fatigue usage factor was calculated to be 0.00136. The fatigue usage factor is less than 1.0.

c. The original design basis stress analysis utilized the NRC approved WECAN computer program to perform a plastic analysis of the divider plate. The design basis plastic analysis was done for any transient cases where stresses exceeded $3S_m$. The applicable transients that were evaluated on a plastic basis were: 1) primary hydrotest, 2) secondary hydrotest, 3) primary leak test, 4) tube leak test (840 psi), 5) tube leak test (600 psi), 6) loss of load, 7) secondary leak test, 8) Reactor Coolant System venting, 9) tube leak test (400 psi), 10) inadvertent Reactor Coolant System depressurization.

From the plastic analysis, the strains, principal strains, and strain range were evaluated and the alternating stress was calculated converting strain to stress. Plastic fatigue usage and elastic fatigue usage were calculated separately and the total fatigue usage was calculated.

The strain concentration factor was used for the fillet at the junction of the divider plate and tubesheet. The factor was applied to the strain component in the Y-direction (ξ yy), principal strains :$\xi 1$, $\xi 2$, $\xi 3$. Strain ranges: $\xi 1$ - $\xi 2$, $\xi 2$ - $\xi 3$, $\xi 3$ - $\xi 1$ were then calculated.

$$S_a \text{ (Alternating Stress)} = \left(\frac{Strainrange}{2}\right) \times [E] \quad \text{where : E = 26.0 x } 10^6 \text{ psi}$$

Having calculated the plastic stresses/strains in the divider plate for the appropriate transients, the plastic fatigue evaluation was done for the original design conditions.

Elastic fatigue usage: Elastic fatigue usage was calculated separately for the transients whose stress intensity range < $3S_m$.

The design basis total fatigue usage = Plastic fatigue + Elastic fatigue (original design condition)

For the SPU case: the evaluation for the SPU conditions was done by applying scale factors to the stresses evaluated in the original design basis stress report. The scale factors were calculated based on the SPU conditions, and then applied to the reference alternating stresses for the affected transients involved in the stress/fatigue evaluation.

The scale factors are based on the ratio of primary side to secondary side differential pressures (ΔP) at the SPU power to the primary to secondary ΔP at the original design conditions (considered in the original design basis stress report).

After applying the scale factors, the revised stress range/fatigue usage was calculated due to SPU conditions.

d. As per ASME Code NB-3228.5: (simplified elastic plastic analysis), The $3S_m$ limit for primary plus secondary stress intensity (NB-3222.2) may be exceeded provided

a. The range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses, shall be $\leq 3S_m$.

b. The value of S_a used for entering the design fatigue curve is multiplied by the factor K_e where

$K_e = 1.0$, for $S_n \leq 3S_m$

$$K_e = 1 + \frac{(1-n)}{n(m-1)}\left\{\left(\frac{S_n}{3S_m} - 1\right)\right\}$$

For $3S_m < S_n < 3mS_m$

S_n = range of primary plus secondary stress intensity, psi

$3S_m$ = limit on the maximum stress range intensity.

n = 0.2 (carbon steel), 0.3 (nickel chromium iron)

m = 3.0 (carbon steel), 1.7 (nickel chromium iron)

The above cited criteria, NB-3228.5, simplified elastic-plastic analysis, was used in the original design basis stress reports for calculating fatigue usage.

For the SPU Case: The SPU evaluation is done by using scale factors. The scale factors are based on the ratio of primary side to secondary side differential pressures (ΔP) at the SPU power to the primary to secondary ΔP at the original design conditions (considered in the design basis stress report). The scale factors were then applied on the reference alternating stresses for the affected transients involved in the stress/fatigue evaluation (stress report) and revised stress range/fatigue usage was calculated for the SPU operating conditions.

e. Two options were used in the evaluation of the tube to tubesheet weld.

Option 1:

Using elastic stress results for all transients regardless of the primary plus secondary stress intensity range, apply the factor 4.0 to the linearized in-plane stresses and the factor 1.215 to the tangential stresses.

Option 2:

For transients creating primary plus secondary stress intensity ranges greater than $3S_m$, compute inelastic strains and modify only the tangential strains by the factor 1.215. Using total strains (not linearized) at the locations of interest, compute the effective uniaxial strain by the equation

$$\epsilon_{eqv} = \frac{\sqrt{2}}{3}\left[(\epsilon_x - \epsilon_y)^2 + (\epsilon_y - \epsilon_z)^2 + (\epsilon_z - \epsilon_x)^2 + \frac{3}{2}(\gamma_{xy}{}^2 + \gamma_{yz}{}^2 + \gamma_{xz}{}^2)\right]^{1/2}$$

Compute a modified Poisson's ratio by the equation

$$\nu = 0.5 - 0.2\,\frac{\epsilon_{yield}}{\epsilon_{eqv}}\text{ ; but not less than } \nu = 0.3$$

Where: ϵ_{yield} = yield strain used in the analysis = 0.000902 in/in

Calculate pseudo-elastic stress components by Hooke's Law

$$\sigma_x = \frac{\nu E(\varepsilon_x + \varepsilon_y + \varepsilon_z)}{(1 + \nu)(1 - 2\nu)} + \frac{E}{(1 + \nu)} \varepsilon_x$$

$$\sigma_y = \frac{\nu E(\varepsilon_x + \varepsilon_y + \varepsilon_z)}{(1 + \nu)(1 - 2\nu)} + \frac{E}{(1 + \nu)} \varepsilon_y$$

$$\sigma_z = \frac{\nu E(\varepsilon_x + \varepsilon_y + \varepsilon_z)}{(1 + \nu)(1 - 2\nu)} + \frac{E}{(1 + \nu)} \varepsilon_z$$

$$\tau_{xy} = G\gamma_{xy}$$

$$\tau_{yz} = G\gamma_{yz}$$

$$\tau_{xz} = G\gamma_{xz}$$

$$G = \frac{E}{2(1 + \nu)}$$

All of the above are based on $E = 26.0 \times 10^6$ lb/in^2 which is consistent with ASME Code Figure I-9.2.

The resulting stresses for transients whose elastically calculated primary plus secondary ranges exceeded $3S_m$ were then combined with all other transients, - calculated as described in Option 1, to compute the fatigue usage.

Option number 1 was the more conservative of the two at the weld surface.

Option number 2 was more conservative at the weld root (Reference Stress Report).

The maximum calculated fatigue usage is at the weld surface. (Option 1)

The WECAN computer code was used in the original design basis analysis stress report to calculate component stresses.

For the SPU operating condition: The scale factors calculated based on the ratio of primary side to secondary side differential pressures (ΔP) at the SPU power to the primary to secondary ΔP at the original design conditions were applied to the reference alternating stresses for the affected transients involved in the stress/fatigue evaluation (stress report). Based on the scaled alternating stresses the revised stress range/fatigue usage for the SPU condition was then calculated.

Example Calculation illustrating the treatment of inelastic strain results (taken from the original design basis stress report):

PRIMARY HYDROTEST INELASTIC STRAIN
MANIPULATION, SECTION 3, WELD ROOT

Parameter	Value
ϵ_x	0.000222 in/in
ϵ_y	0.0020505 in/in
ϵ_z'	-0.0033755 in/in
$\epsilon_z = 1.215\ \epsilon_z'$	-0.0041012 in/in
γ_{xy}	0.004037 in/in
γ_{yz}	0.0 in/in
γ_{xz}	0.0 in/in
ϵ_{eqv}	0.0043288 in/in
ν	0.4583
σ_x	-175.3 ksi
σ_y	-142.7 ksi
σ_z	-252.4 ksi
τ_{xy}	36.0 ksi
τ_{yz}	0.0 ksi
τ_{xz}	0.0 ksi

RAI #24

In Section 5.2.7, Structural Evaluation of Reactor Internal Components," FPLE states the following:

> ... the reactor pressure vessel internals were designed to meet the intent of Section III, Subsection NG of the ASME Boiler and Pressure Vessel Code. Plant-specific stress report on the reactor pressure vessel internals was not required. The structural integrity of the Seabrook Station reactor pressure vessel internals design has been ensured by analyses performed on both generic and plant-specific bases.

Provide a comparison of the calculated stresses to the allowed stresses of Subsection NG of the ASME Code.

FPL Energy Seabrook Response:

The following Tables RAI 24-1 and RAI 24-2 give the calculated stresses versus allowable stresses for some critical components of the reactor internals for Seabrook Station. Westinghouse proprietary information is provided in Enclosure 2.

TABLE RAI 24-1
Most Critical Reactor Internal Components Calculated Stresses Allowable Stresses and Fatigue Usage (Most Critical Section)

Component	Max Stress (P_m) [psi]	Code Limit (S_m) [psi]	Max Stress (P_m+P_b) [psi]	Code Limit (1.5S_m) [psi]	Max Stress (P_m+P_b+Q) [psi]	Code Limit (3S_m) [psi]	Fatigue (U)
Lower Support Columns	[ ]a,c	16,100	[ ]a,c	24,150	[ ]a,c	48,300	[ ]a,c
Core Barrel Nozzle	[ ]a,c	16,400	---	---	[ ]a,c	49,200	[ ]a,c

TABLE RAI 24-2
Summary of Results for Core Plates (Most Critical Section)

Component	Category	Maximum Stress Value (psi)	Allowable Stress Value (psi)	Margin of Safety	Cumulative Fatigue Usage
Lower Core Plate	$P_m + P_b + Q$	[ ]a,c	48,600	[ ]a,c	[ ]a,c
Upper Core Plate	$P_m + P_b + Q$	[ ]a,c	48.6	[ ]a,c	[ ]a,c

RAI #25

In Section 5.4.3, "Description of Analyses and Evalutions, FPLE discusses the evaluation of the CRDMs and states the following:

> The only evaluations that were not bounded were those associated with the changes in NSSS design transients that were not enveloped by the current analyses. Ratios of the new transients to the old transients were used (very small change, less than 5%) to multiply the existing stress evaluation results. After this was performed, it was shown that the component stresses were within the allowable limits of the ASME Boiler and Pressure Vessel Code.

However, the application does not provide the new stresses or the margin to allowable of the current stresses. Provide the above information to support the assertion that the new stresses are acceptable.

FPL Energy Seabrook Response:

Results are shown in Table RAI 25-1 for calculated versus allowable stresses for various components of the control rod drive mechanisms. Westinghouse proprietary information is provided in Enclosure 2.

TABLE RAI 25-1

		Design Condition		Normal Condition		Upset Condition		Testing Condition		Faulted Condition		Emergency Cond.	
Component	Parameters Per ASME Code III	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)
UPPER JOINTS													
Cap	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Rod Travel Housing	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.000	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Canopy	Pm	[ ]^{a,c}	16.900	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	N/A		N/A		Note 1		N/A		Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Threaded Area Joints	Pm	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660
	Stress Intensity Due to Bell Mouth	N/A		[ ]^{a,c}	17.900	[ ]^{a,c}	17.900	[ ]^{a,c}	17.900	N/A		N/A	

TABLE RAI 25-1

		Design Condition		Normal Condition		Upset Condition		Testing Condition		Faulted Condition		Emergency Cond.	
Component	Parameters Per ASME Code III	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)	Calculated (ksi)	Allowed (ksi)
MIDDLE JOINT													
Rod Travel Housing	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	N/A		[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Latch Housing	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	N/A		[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Canopy	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	N/A		N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	80.000	[ ]^{a,c}	80.000	[ ]^{a,c}	80.000	[ ]^{a,c}	80.000	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Threaded Area Middle Joints	Pm	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	N/A		[ ]^{a,c}	9.660
	Stress Intensity Due to Bell Mouth	N/A		[ ]^{a,c}	17.900	[ ]^{a,c}	17.900	[ ]^{a,c}	17.900	N/A		N/A	

TABLE RAI 25-1

		Design Condition		Normal Condition		Upset Condition		Testing Condition		Faulted Condition		Emergency Cond.	
Component	**Parameters Per ASME Code III**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**
LOWER JOINTS													
Latch Housing	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Head Adapter	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Canopy	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	N/A		N/A		Note 1		N/A		Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Threaded Area Lower Joints	Pm	[ ]^{a,c}	9.660	N/A		[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	N/A		[ ]^{a,c}	9.660
	Stress Intensity Due to Bell Mouth	N/A		[ ]^{a,c}	20.700	[ ]^{a,c}	20.700	[ ]^{a,c}	20.700	N/A		N/A	

TABLE RAI 25-1

		Design Condition		Normal Condition		Upset Condition		Testing Condition		Faulted Condition		Emergency Cond.	
Component	**Parameters Per ASME Code III**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**	**Calculated (ksi)**	**Allowed (ksi)**
CAPPED LATCH HOUSING													
Cap	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Latch Housing	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	[ ]^{a,c}	24.150	N/A		Note 1		[ ]^{a,c}	40.500	Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	48.300	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Canopy	Pm	[ ]^{a,c}	16.100	N/A		Note 1		[ ]^{a,c}	27.000	[ ]^{a,c}	38.640	Note 1	
	Pm + Pb	N/A		N/A		Note 1		N/A		Note 1		Note 1	
	Pm+Pb+Q	N/A		[ ]^{a,c}	48.300	[ ]^{a,c}	50.700	N/A		N/A		N/A	
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	64.400	[ ]^{a,c}	154.560	[ ]^{a,c}	64.400
Threaded Area CLH	Pm	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	[ ]^{a,c}	9.660	N/A		[ ]^{a,c}	9.660
	Stress Intensity Due to Bell Mouth	N/A		[ ]^{a,c}	17.900	[ ]^{a,c}	17.900	[ ]^{a,c}	17.900	N/A		N/A	

Note 1: Previously analyzed loads remain bounding.

RAI #26

In Section 5.6.2, "Results," FPLE describes the results for the pressurizer surge line and states the following:

The design basis analysis for the Seabrook Station pressurizer surge nozzle did however, originally show the cumulative fatigue usage factor to be close to 1.0 prior to consideration of the SPU conditions. For the surge nozzle, a comparative evaluation was performed utilizing stress and fatigue results from another unit having essentially the same pressurizer design. The comparison identified significant conservatisms in the method used for design transient combinations for Seabrook Station. By adopting the evaluation from the comparison unit, it was demonstrated that the more accurate analytical effort resulted in acceptable fatigue usage for operating conditions which envelop the Seabrook Station SPU conditions.

The application does not provide a clear or complete description of what was changed in the analyses or why the analysis from the comparison unit is appropriate for Seabrook. Provide the above information in order to justify adopting the analysis of the comparison unit.

FPL Energy Seabrook Response:

The original design basis analysis for Seabrook Station shows that the cumulative fatigue usage for the surge line nozzle is close to 1.0 prior to the consideration of the SPU conditions. Review of the original design transient conditions analyzed for the pressurizer (and the surge nozzle) confirmed that the transients in the current analysis of record bound the design transients at the SPU conditions.

The current design basis analysis for the Seabrook Station, however, does not require that the impact of thermal stratification on the pressurizer surge line nozzle be addressed. As a conservative measure, a comparative analysis was performed for the SPU, using the design analysis from Millstone Unit 3, to estimate the impact of thermal stratification on the Seabrook Station pressurizer surge line nozzle cumulative fatigue usage factor.

Millstone Unit 3 utilizes the same pressurizer model as Seabrook Station, and design qualification for both units is based on the same stress report. For the Millstone Unit 3 pressurizer, more extensive analyses (to supplement the original stress report results) were performed for the surge nozzle, which is the same geometry and material as Seabrook Station. The additional analyses performed for the Millstone Unit 3 surge nozzle addressed thermal stratification and removed excessive conservatism from the original design basis. Since the geometry (i.e., critical dimensions), materials, and the ASME Code of record are the same for both the Seabrook Station and Millstone Unit 3, the analysis for Millstone Unit 3 can be utilized to assess the effect of consideration of thermal stratification on the Seabrook Station design analysis.

A fatigue usage value of nearly 1.0 was calculated in the original design basis analysis for the surge line nozzle for Millstone 3. The more detailed analysis for thermal stratification for Millstone 3 resulted in the calculation of a significantly lower fatigue usage for the surge nozzle (approximately 0.3) considering the surge line stratification effects. The reduction in fatigue usage is due to unrealistic and overly conservative modeling of the steady state temperature fluctuations and thermal stresses on the surge nozzle during certain heatup and cooldown transients.

Based on the comparative analysis, it is expected that the Seabrook Station pressurizer surge line nozzle cumulative fatigue usage factor would be less than 1.0, even with consideration of thermal stratification.

Based on the above, the impact of operation of the Seabrook Station pressurizer at SPU conditions as bounded by the original design basis analysis remains below the ASME Code acceptance limit of 1.0. In addition, when the effects of thermal stratification are included it is also expected that the fatigue usage factor would be less than 1.0. As a result, the current design analysis remains the Current Analysis of Record.

RAI #27

In Section 5.8.1.3.2, "Transient Evaluation," FPLE states the following:

> The original qualification of the pump was based on using a fatigue waiver, as defined in Section NB-3222.4(d) of the ASME Boiler and Pressure Vessel Code, to address fatigue for the Code pressure boundary parts of the pump. The revised calculations show that the pump casing, the thermal barrier, bolting ring and main flange bolts, and the seal housing still qualify for the fatigue waiver.

Provide stresses and cumulative usage factors for the indicated reactor coolant pump (RCP) components for the SPU condition. Discuss how these components qualify for the fatigue waiver.

FPL Energy Seabrook Response:

Per Section NB-3222.4(d) of the ASME Code, an analysis for cyclic operation is not required, and it may be assumed that the peak stress limit discussed in NB-3222.4(b) is satisfied, if the specified normal conditions (which include upset conditions) of the component or portion thereof meet the six conditions stipulated in NB-3222.4(d) (1) through (6). The Seabrook Station reactor coolant pump casing, thermal barrier, bolting ring, and seal housing meet these six conditions as discussed below:

Condition 1 – Atmospheric to Operating Pressure Transients

This evaluation assesses the ability of the reactor coolant pump components (casing, thermal barrier, seal housing, and bolting ring) to withstand the number of pressure cycles associated with large ΔPs that result from exposing the components from atmospheric pressure to plant operating pressure. For this assessment, the following steps are taken:

- Summarize large ΔP transients
- Estimate the number of cycles each component can withstand
- Compare the allowable transients of step 2 above against the requirement of step 1 above.

The full pressure (large ΔP) cycles* are:

Heatup/Cooldown	200 cycles
Inadvertent Reactor Coolant System Depressurization	20 cycles
Primary Side Leak Test	200 cycles
Secondary Side Leak Test	80 cycles
Turbine Roll Test	20 cycles
Primary Side Hydrostatic Test	10 cycles
Total =	530 cycles

*All cycles listed are consistent with the Seabrook Station UFSAR design basis cycles

The number of allowable cycles for each component is determined from the alternating stress S_a that is equal to $3S_m$ on the applicable fatigue curve in Figures I-9.0 of the ASME Code, where the S_m (the stress intensity) is based on the component operating temperature. Table RAI 27-1 below summarizes the calculation of the allowable number of atmospheric-to-operating pressure cycles.

TABLE RAI 27-1

Component	Temperature (°F)	S_m (ksi)	$3S_m$ (ksi)	N^* (cycles)	N > 530 cycles?
Casing	557	15.9	47.81	28,000	Yes
Thermal Barrier	408	18.6	55.79	12,000	Yes
Seal Housing	131	20.0	60.0	9,200	Yes
Retaining Ring	187	26.7	80.1	1,200	Yes

* From ASME Code Section III Figure I-9.0 curves

Based on the results of Table RAI 27-1, all the reactor coolant pump components considered have allowable large ΔP cycles that exceed 530 cycles.

Condition 2 – Normal Operating Pressure Fluctuations

This evaluation confirms that the normal pressure fluctuations associated with the SPU meet the ASME Code requirements in NB-3222.4(d) (2). In accordance with the ASME Code requirements, one of the conditions for waiving the fatigue evaluation is if the specified full range of pressure fluctuations during normal operation does not exceed the quantity 1/3 x design pressure x (S_a/S_m), where S_a is the value obtained from the applicable design fatigue curve for the total specified number of significant pressure fluctuations and S_m is the allowable stress intensity for the material at operating temperature. If the total specified number of significant pressure fluctuations exceeds 10^6, the S_a value at $N = 10^6$ may be used. Significant pressure fluctuations are those for which the total excursion exceeds the quantity:

$$Ps = \text{Design Pressure} \times \frac{1}{3} \times \frac{S}{S_m}$$

Where:

Design Pressure = 2500 psi

Ps = Pressure above which a pressure fluctuation is significant

$S = S_a$ at 10^6 cycles

S_m = Allowable stress intensity of the component at the operating temperature

S_a = Alternating stress from the ASME Code Figures I-9.0

The pressure above which a pressure fluctuation is significant for each component is calculated below:

TABLE RAI 27-2

Component	Sm (ksi)	S (ksi)	Ps (psi)
Casing	15.94	26.0	1359
Thermal Barrier	18.6	26.0	1165
Seal Housing	20.0	26.0	1083
Bolting Ring	26.7	13.0	406

The significant pressure cycles* for the casing, thermal barrier, and seal housing are:

Heatup/Cooldown	200 cycles
Inadvertent Reactor Coolant System Depressurization	20 cycles
Primary Side Leak Test	200 cycles
Secondary Side Leak Test	80 cycles
Primary Side Hydrostatic Test	10 cycles
Total =	510 cycles

The significant pressure cycles* for the bolting ring are:

Heatup/Cooldown	200 cycles
Loss of Load	80 cycles
Loss of Power	40 cycles
Reactor Trip, cooldown, no Safety Injection	160 cycles
Reactor Trip, cooldown and Safety Injection	10 cycles
Inadvertent Reactor Coolant System Depressurization	20 cycles
Excessive Feedwater Flow	30 cycles
Turbine Roll Test	20 cycles
Primary Side Leak Test	200 cycles
Secondary Side Leak Test	80 cycles
Primary Side Hydrostatic Test	10 cycles
Total =	850 cycles

*All cycles listed are consistent with the Seabrook Station UFSAR design basis cycles

The allowable full range of pressure fluctuations is then calculated as follows:

TABLE RAI 27-3

Component	P_s (psi)	N_s, cycles	S_a (ksi)	P_r (psi)
Casing	1359	510	135	7058
Thermal Barrier	1165	510	135	6048
Seal Housing	1083	510	135	5625
Retaining Ring	406	850	87.7	2737

Where:

Ps = Pressure above which a pressure fluctuation is significant, calculated previously based on S_a, and S_m at design pressure and temperature.

N_s = Number of significant normal operation pressure fluctuations

S_a = Alternating stress from the ASME Code for the number of significant normal operation pressure fluctuations.

P_r = Allowable full range of pressure fluctuations

Since none of the normal operating transients has pressure fluctuations as large as the calculated allowable full range of pressure fluctuations, this requirement is satisfied for the casing, thermal barrier, seal housing, and retaining ring.

Condition 3 – Temperature Difference – Startup and Shutdown

The evaluation is to assess whether or not the temperature difference, in °F, between any two adjacent points of the component during normal operation exceeds the temperature difference arising from a limiting stress level S_a associated with the number of startup and shutdown cycles. The limiting temperature difference is calculated as follows:

$$\Delta T_a = S_a/2E\alpha$$

where S_a is the alternating stress based on the number of startup and shutdown cycles. The Young's Modulus (E) and the coefficient of thermal expansion (α) are based on the component material mean temperature (T_m). This evaluation was carried out in the table below. The final column of the table, ΔT_o, gives the maximum calculated temperature difference for the startup and shutdown events from the original calculations. There are no temperature changes between the original design transients and the SPU transients with regard to startup and shutdown events. Therefore, the Condition 3 assessment is satisfied.

TABLE RAI 27-4

Component	T_m (°F)	α (10^{-6}/°F)	E (10^6 psi)	S_a (ksi)	ΔT_a (°F)	ΔT_o (°F)
Casing	338	9.84	26.91	186	351.2	295
Thermal Barrier	264	9.68	27.32	186	351.7	198
Seal Housing	126	9.30	28.04	186	356.6	41
Bolting Ring	154	6.46	29.64	160	417.8	30

Condition 4 – Temperature Difference – Normal Operation

This evaluation is to show that the maximum temperature difference in °F between any two adjacent points does not change during normal operation by more than $S_a/2E\alpha$, where S_a is the value obtained from the applicable design fatigue curve of Figure I-9.0 of the ASME Code for the total number of significant temperature fluctuations. A temperature difference fluctuation is defined as significant if its total algebraic range exceeds the quantity $S/2E\alpha$, where S is the value of S_a obtained from the applicable design fatigue curve for 10^6 cycles. The calculation of the significant temperature fluctuation is shown below, using maximum component temperatures:

TABLE RAI 27-5

Component	T_m (°F)	α (10^{-6}/°F)	E (10^6 psi)	S (ksi)	ΔT_s (°F)
Casing	557	10.39	25.7	26	48.7
Thermal Barrier	408	9.69	26.6	26	49.0
Seal Housing	131	9.32	28.0	26	49.8
Bolting Ring	187	6.61	29.6	13	33.2

The number of significant temperature fluctuations that exceed the above ΔT_S values is listed below*:

Turbine Roll Test	20 cycles
Loss of Load	80 cycles
Loss of Power (Bolting Ring Only)	40 cycles
Reactor Trip, cooldown and Safety Injection Reactor Trip, cooldown and Safety Injection	10 cycles
Inadvertent Reactor Coolant System Depressurization	20 cycles
Excessive Feedwater Flow	30 cycles
Total for bolting ring	= 200 cycles
Total for other components	= 160 cycles

*All cycles listed are consistent with the Seabrook Station UFSAR design basis cycles

The allowable temperature differences are calculated using the equation $\Delta T_d = S_a/2E\alpha$. The results of the evaluation are shown below:

TABLE RAI 27-6

Component	α, (10^{-6}/°F)	E, (10^6 psi)	N, cycles	S_a, (ksi)	ΔT_d, (°F)	ΔT_o (°F)
Casing	10.39	25.7	160	200	374	295
Thermal Barrier	9.98	26.6	160	200	377	198
Seal Housing	9.32	28.0	160	200	383	41
Bolting Ring	6.61	29.6	200	155	396	30

The final column of the table, ΔT_o, gives the maximum calculated temperature differences. Since these are all less than the ΔT_d values, the Condition 4 assessment is satisfied.

Condition 5 – Temperature Difference– Dissimilar Materials

For a component fabricated from dissimilar materials, the difference in E and α can cause a stress due to uniform heatup. This difference does not apply to the reactor coolant pump pressure boundary components, so this condition is satisfied.

Condition 6 – Mechanical Loads

This evaluation covers the effect of externally induced loads on the reactor coolant pump components due to the transients. The full range of mechanical loads (excluding pressure) including pipe reactions shall not result in stresses whose range exceeds the Sa value for the total significant load fluctuation associated with the transients. For the reactor coolant pump, the only imposed loads are the support feet and nozzle seismic loads. The SPU requirements have not introduced new seismic loads. Assuming that the seismic loads consisting of 400 cycles qualify as significant loads, the Sa values per Figures I-9.0 of the ASME Code would be 120,000 psi for the carbon steel bolting ring and 146,000 psi for the other stainless steel components. The design base cycles listed in the Seabrook Station UFSAR lists the 5 earthquakes of 10 cycles each are to be considered, this analysis was performed consistent with the Seabrook Station reactor coolant pump design specification of 20 operating basis earthquake (OBE) seismic occurrences of 20 cycles each. Since normal/upset stress limits are less than these values, this condition is met.

Fatigue Waiver for Main Flange Bolts

The fatigue analysis requirements for the main flange bolts are given in Section NB-3232.3 of the ASME Code. This section allows the fatigue waiver to apply to bolts when the components on which they are installed meet all of the conditions of NB-3222.4 (d) and thus require no fatigue analysis. The main flange bolts are installed through the bolting ring and into tapped holes in the casing. The bolts also capture the thermal barrier flange for the joint between it and the casing. Since all of these components meet the fatigue waiver requirements of NB-3444.4(d), the fatigue waiver also applies to the main flange bolts.

Component Stresses and Cumulative Usage Factors

No new stresses or usage factors were calculated for the SPU conditions, as the existing analyses were found to continue to apply for the SPU conditions. The following stresses and usage factors are taken from previous analyses.

TABLE RAI 27-7
Design Condition - Non-Bolting Components

Component	P_m (psi)		$(P_L \text{ or } P_m) + P_b$ (psi)	
	Calculated Value	Allowable Value	Calculated Value	Allowable Value
Bolting Ring	<20,355	26,700	<20,355	40,050
Seal Housing	<15,102	20,000	<15,240	30,000
Thermal Barrier Flange	13,524	18,050	24,659	27,075
Casing	14,075	15,300	17,365	22,950
Suction/Discharge Nozzles	15,250	15,300	20,555	22,950

TABLE RAI 27-8
Normal/Upset Conditions - Non-Bolting Components

Component	$(P_L \text{ or } P_m) + P_b + Q$ (psi)		Fatigue Usage	
	Calculated Value	Allowable Value	Calculated Value	Allowable Value
Bolting Ring	17,897	80,100	Fatigue Waiver	–
Seal Housing	<14,864	60,000	Fatigue Waiver	–
Thermal Barrier Flange	44,373	60,050	Fatigue Waiver 0.575 (Note 1)	– 1.0
Casing (Note 2)	61,988	54,300	0.029	1.0
Suction/Discharge Nozzles (Note 2)	54,285	49590	0.029	1.0

Note 1: The fatigue waiver is demonstrated in the text for the primary system transients. A fatigue analysis is performed for the inside of the hole in the thermal barrier flange where component cooling water is supplied to the thermal barrier heat exchanger, for the auxiliary system transients defined for the cooling water nozzles. The usage factor given is from an analysis performed in 2003 to support the use of 35°F component cooling water.

Note 2: Acceptability of the casing stress range and the suction and discharge nozzles stress range is demonstrated by use of a simplified elastic-plastic analysis per Section NB-3228.3 of the ASME Code. The usage factors calculated are from those analyses.

TABLE RAI 27-9
Main Flange Bolts

Condition	Type of Stress	Calculated Value	Allowable Value
Design	P_m	31,996	33,580
Normal	P_m	46,762	73,280
Normal	$P_L + P_b$	76,115	109,920
Upset	P_m	47,366	70,920
Upset	$P_L + P_b$	70,612	106,380
Normal/Upset	Cumulative Usage Factor	Fatigue Waiver	–

RAI #28

In Section 9.2.2.1, "Inside Containment," FPLE states the following:

> ... the protective coatings (i.e., organic material) will continue to meet the requirements of Regulatory Guide 1.54 [Reference 9.2-3] and will be acceptable following implementation of the SPU.

This statement is unclear. Provided an explanation of what is meant by this statement.

FPL Energy Seabrook Response:

As a result of the SPU, the integrated radiation dose (40-year normal operation plus accident dose) will increase in certain areas of the containment, and therefore an evaluation of the qualified protective coatings inside containment was performed. Note that the normal inside containment dose did not change as a result of SPU. Qualified containment coating systems are tested to 1.0 E09 rads. The total integrated dose inside containment at current conditions is 2.0 E08 rads and at SPU conditions is 2.1 E08 rads. Therefore, the SPU integrated dose is still within the dose qualification of the current qualified coatings.

Fire Protection Engineering

RAI #29

NRC RS-001, Rev. 0, "Review Standard for Extended Power Uprates," Attachment 2 to Matrix 5, "Supplemental Fire Protection Review Criteria, Plant Systems," states that "... power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's application should confirm that these elements are not impacted by the extended power uprate ..."

Section 9.1.1, "Fire Protection Program," of the March 17, 2004, submittal does not address items (1) through (5) above. Provide statements to address these items.

FPL Energy Seabrook Response:

Response is based on Attachment 1, to Matrix 5 of RS-001, in lieu of Attachment 2 as referenced above.

As presented in LAR Attachment 1, Subsection 4.1.4.3.3 "Appendix R and Safe Shutdown Cooldown" (page 4-8) cooldown will not change as a result of SPU, i.e., cold shutdown is achieved and maintained within 72 hours. The updated cooldown analysis addressing SPU confirms that cold shutdown can be achieved and maintained using this one train, inclusive of the additional burden associated with SPU. Appendix R program administrative controls, as well as the elements of the program, including fire suppression; fire barriers; and fire protection responsibilities of plant personnel are unchanged. Procedures and resources necessary for the repair of systems required for achieving and maintaining cold shutdown are unaffected, and the potential for a radiological release following a fire is also unchanged.

RAI #30

NRR RS-001, Rev. 0, Attachment 2 to Matrix 5, states that "... where licensees rely on less than full capability systems for fire events ..., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability ..."

Licensees should identify the impact of the SPU on a plant's post-fire safe-shutdown procedures. Sections 4.1.4.3.3, "Appendix R and Safe Shutdown Cooldown," and 9.1.1, of the March 17, 2004, submittal do not address items (1) and (2) above. Provide statements to address these items.

FPL Energy Seabrook Response:

Response is based on Attachment 1, to Matrix 5 of RS-001, in lieu of Attachment 2 as referenced above.

Seabrook Station does not rely on less than full capability systems for fire events. Seabrook Station meets the requirements of 10 CFR 50 Appendix R using a single train of systems and components at SPU conditions. This complies with the current licensing basis for Seabrook Station, which is described in UFSAR Appendix R Section 3.1, "Safe Shutdown Capability". There are no modifications necessary to the Fire Protection System or Fire Protection Program as a result of SPU. Additionally, there is no impact on the Seabrook Station post-fire safe shutdown procedures.

PWR Systems

RAI #31

Table 1.2-1 of the March 17, 2004, submittal, lists the computer codes used to perform non-LOCA [loss of coolant accident] transient and accident reanalyses for the SPU at Seabrook. RETRAN-02 has been generically approved by the NRC staff for non-LOCA transient analyses; however, this is the first time RETRAN-02 will be used at Seabrook. Explain the quality assurance process used to verify that RETRAN-02 was adequately used at Seabrook and show that the Seabrook nodalization modeling is consistent with the Westinghouse 4-loop plant nodalization model of WCAP-14882-P-A. If the modeling of Seabrook deviated from the plant model in the WCAP-14882-P-A, justify the technical validity. Identify and explain which accident analyses the use of RETRAN code is not suitable.

FPL Energy Seabrook Response:

1. The Westinghouse Quality Assurance Program computer software development, maintenance and configuration control process is in accordance with procedures and instructions that comply with ASME NQA-1 and International Organization for Standardization Standard ISO 9001 "Quality Management Systems – Requirements" for all safety-related applications.

   The RETRAN-02 computer code approved for use in performing Westinghouse safety analyses (WCAP-14882-P-A) is validated and documented under the Westinghouse software configuration control process governed by the NRC-Approved Westinghouse Quality Management System.

   For the Seabrook Station, non-LOCA safety analyses performed with RETRAN, Westinghouse utilized internal methodology guidance documents known as RETRAN Safety Analysis Standards to ensure that the assumed initial conditions and other plant parameters are conservative, and that applicable safety evaluation report requirements are satisfied. Each RETRAN Safety Analysis Standards document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions, e.g., directions of conservatism for initial condition values, a detailed description of the transient model development, and a discussion of the expected transient analysis results. Finally, independent verification of each event analysis confirmed that approved versions of the applicable code and RETRAN Safety Analysis Standards were appropriately applied.

2. The RETRAN nodalization modeling used in the Seabrook Station analyses is consistent with the Westinghouse 4-loop plant nodalization model of WCAP-14882-P-A. Although not considered to be an inconsistency with the model of WCAP-14882-P-A, the Seabrook Station model has each hot leg divided into three equal control volumes instead of one. This minor model enhancement was made to minimize code instabilities attributed to pressurizer insurge and outsurge; although only needed for the hot leg connected to the pressurizer, all loops were divided the same. To ensure the approved RETRAN model is applied in each analysis, Westinghouse used a pre-processor computer code to generate the base RETRAN input model. Like RETRAN, the pre-processor computer code is

validated and documented under the Westinghouse software configuration control process governed by the NRC-Approved Westinghouse Quality Management System.

3. Consistent with Table 1 of the NRC Safety Evaluation Report for WCAP-14882-P-A, Westinghouse limits the use of the RETRAN code to the following transients: feedwater system malfunctions, excessive increase in steam flow, inadvertent opening of a steam generator relief or safety valve, steam line break, loss of external load/turbine trip, loss of offsite power, loss of normal feedwater, feedwater line rupture, loss of forced reactor coolant flow, locked reactor coolant pump rotor/sheared shaft, control rod cluster withdrawal at power, dropped control rod cluster/dropped control bank, inadvertent increase in coolant inventory, inadvertent opening of a pressurizer relief or safety valve, and steam generator tube rupture.

   The transients analyzed for the Seabrook Station using RETRAN are as follows: feedwater system malfunction (UFSAR 15.1.2), steam line break (UFSAR 15.1.5), loss of external electrical load/turbine trip (UFSAR 15.2.2, 15.2.3), loss of normal feedwater (UFSAR 15.2.7), loss of AC power to the plant auxiliaries (UFSAR 15.2.6), feedwater system pipe break (UFSAR 15.2.8), loss of reactor coolant flow (UFSAR 15.3.1, 15.3.2), locked rotor/shaft break (UFSAR 15.3.3, 15.3.4), uncontrolled rod cluster control assembly withdrawal at power (UFSAR 15.4.2), inadvertent operation of emergency core cooling system during power operation (UFSAR 15.5.1), Chemical and Volume Control System malfunction that increases reactor coolant inventory (UFSAR 15.5.2), inadvertent opening of a pressurizer safety or relief valve (UFSAR 15.6.1), and steam generator tube rupture (UFSAR 15.6.3). As noted in Table RAI 43-2 each of these transients corresponds to one of the transients listed in Table 1 of the Safety Evaluation Report.

RAI #32

Table 2.3-1 lists the design operating parameters. Provide a tabulation of the thermal design parameters compared to values assumed in the safety analyses to demonstrate that adequate conservatism is available for the safety analyses assumptions.

Table 6.3.1-2 lists non-LOCA plant initial conditions assumptions for the SPU. Explain why these parameters values are same as Table 2.3-1 which shows design operating parameters. In this table, explain why the initial power condition uncertainty is ± 0% rated thermal power.

Discuss the basis for the average temperature operating range (571.0°F - 589.1°F) listed in Table 2.3-1.

FPL Energy Seabrook Response:

1. Table RAI 32-1 provides a comparison between thermal design parameter values and the corresponding safety analysis values for the Seabrook Station SPU. With the exception of the NSSS and core power, this table clearly demonstrates that conservatism was applied in the safety analyses. The license amendment request includes an SPU core power of 3587 MWt (from 3411 MWt), which is bounded by the core power assumed in the safety analyses, 3659 MWt. The difference of 2% is used to account for the steady-state initial condition uncertainties.

TABLE RAI 32-1

THERMAL DESIGN PARAMETER VALUES VS. SAFETY ANALYSIS VALUES

Parameter	Expected Operating Value	Thermal Design Value	Safety Analysis Value[1]
NSSS power (MWt)	3606	3678	3678 ±0
Core power (MWt)	3587	3659	3659 ±0
Vessel T_{avg}, (°F)	589.1	571.0 (min) 589.1 (max)	571.0 +6/-5 589.1 +6/-5[2]
Pressurizer pressure (psia)	2250	2250	2250 ±50
RCS flow (gpm) Thermal Design Flow Minimum Measured Flow	402,000	374,400 (TDF)	374,400 383,800
Core Bypass (%)	7.0[3]	8.3	8.3 (design) 6.8 (statistical)

NOTES:

(1) For non-LOCA analyses in which the standard thermal design procedure method is applied, the thermal design flow and design core bypass are assumed with initial condition values that include uncertainties. For revised thermal design procedure, departure from nucleate boiling analyses, minimum measured flow and the statistical core bypass are assumed with nominal initial condition values; the initial condition uncertainties are combined statistically with other parameters to define the departure from nucleate boiling ratio safety analysis limit.

(2) For the revised thermal design procedure, departure from nucleate boiling analyses, a random uncertainty of ±3°F was used to define the departure from nucleate boiling ratio safety analysis limit, and a bias of -3°F was applied as a departure from nucleate boiling ratio penalty.

(3) Best estimate value with thimble plugs removed

2. LAR Attachment 1, Table 6.3.1-2 (page 6-58) presents nominal plant conditions, which are the same as the design values in LAR Attachment 1, Table 2.3-1 (page 2-3), and initial condition uncertainties. Combining the nominal plant condition values with the initial condition uncertainties provides the non-LOCA plant initial condition assumptions for the SPU. As for the 0% core power uncertainty, note that a core power of 3659 MWt is assumed in the non-LOCA analyses, consistent with the nominal design core power, to bound the SPU core power of 3587 MWt. The difference of 2% is used to account for the steady-state initial condition uncertainties.
3. The basis for the average temperature operating range is operating flexibility as described in paragraph 2 of LAR Attachment 1, Section 2.1 "Introduction and Background" (page 2-1). An average temperature operating range has been implemented at many Westinghouse plants including Byron 1 and 2, Braidwood 1 and 2, South Texas 1 and 2, and D. C. Cook 1 and 2.

RAI #33

Table 2.3-1, Note 4, discusses core bypass flow which includes 2% for thimble plug removal and 0.5% for intermediate flow mixers. Explain why 0.5% for intermediate flow mixers is not reflected in the table. Should the core bypass flow be 2.5%?

FPL Energy Seabrook Response:

The core bypass flow includes 2% for thimble plug removal. The reference to 0.5% for intermediate flow mixers (IFMs) is deleted from Note 4 of LAR Attachment 1, Table 2.3-1 (page 2-3).

The core bypass flows shown in LAR Attachment 1, Table 2.3-1 are correct.

RAI #34

Table 3.1-1 lists the summary of reactor coolant system (RCS) design transients. Expand this table to include the numbers of cycles for each design transient at the current and SPU conditions. Describe the basis for each of the changes.

FPL Energy Seabrook Response:

Table 3.9-(N)-1 of the Seabrook Station UFSAR lists the Reactor Coolant System design transients for Seabrook Station. The number of cycles for each transient are not changed by the SPU.

RAI #35

The March 17, 2004, submittal provides a summary of LOCA analysis parameters. This submittal also refers to information specific to the LOCA analyses performed to define the licensing basis for Seabrook LOCA. The NRC staff requests further information to address the programmatic requirements of 10 CFR 50.46 (c).

To show that the referenced generically-approved Seabrook small break LOCA (SBLOCA) analysis methodology continues to apply specifically to the Seabrook plant, provide a statement that FPLE and its vendors have ongoing processes that assure the ranges and values of the input parameters for the Seabrook SBLOCA analysis bound the ranges and values of the as-operated plant parameters. Furthermore, if the Seabrook plant-specific analyses are based on the model and/or analyses of any other plant, then justify that the model or analyses apply to Seabrook (e.g. if the other design has a different vessel internals design the model would not apply to Seabrook).

FPL Energy Seabrook Response:

Both Seabrook Station and its LOCA analysis vendor (Westinghouse) have ongoing processes which assure that the values and ranges of the LOCA analyses inputs for peak cladding temperature-sensitive parameters conservatively bound the values and ranges of the as-operated plant for those parameters. Furthermore, Seabrook Station plant-specific LOCA analyses are based on Seabrook specific models.

RAI #36

The LOCA submittals did not address slot breaks at the top and side of the pipe. Justify why these breaks are not considered for the Seabrook large break LOCA (LBSOCA) and SBLOCA responses.

FPL Energy Seabrook Response:

Break location, type, and size are specifically considered for the Seabrook Station LBLOCA analysis. The analysis concluded that the cold leg guillotine break is limiting for Seabrook Station. The uncertainties related to break location, type, and size were included in the model uncertainties for the Seabrook Station Best Estimate LBLOCA peak cladding temperature.

For Small Break LOCA events, the effects of break location have been generically evaluated as part of the application of the NOTRUMP Evaluation Model (Reference 1). This document concluded that a break in the Reactor Coolant System cold leg was limiting. Additionally, the effects of break orientation were considered during the evaluation of safety injection in the broken loop and application of the COSI Condensation Model (Reference 2). This work concluded that a break oriented at the bottom of the Reactor Coolant System cold leg piping was limiting with respect to peak cladding temperature.

While these references specifically address the short-term response to the LOCA break spectrum, the long-term effects associated with potential reactor coolant pump loop seal re-plugging core uncovery is addressed in the following.

A review of the analysis conditions associated with potential core uncovery due to loop seal re-plugging has previously been performed in Reference 3. Reference 3 documents the Westinghouse position with regards to the potential for Inadequate Core Cooling scenarios following large and intermediate break LOCAs as a result of loop seal re-plugging. Reference 3 concludes the following:

- The Reactor Coolant System response following a LOCA is a dynamic process and that the expected response in the long-term is similar to the response that occurs in the short-term. This short-term response has been analyzed extensively through computer analysis and tests and is well documented.
- Consideration of the physical mechanisms for liquid plugging of the pump suction leg U-bend piping following large and intermediate break LOCA at realistic decay heat levels precludes quasi steady-state inadequate core cooling conditions.
- It is important to emphasize that the operator guidance provided in the Emergency Response Guidelines includes actions to be taken in the event of an indication of a challenge to adequate core cooling following a LOCA.

A review of the key contributors associated with long-term loop seal plugging core uncovery scenarios, under LOCA conditions, was performed as part of Reference 4 including a review of pertinent experimental data.

a,c

From References 3 and 4, it can be concluded that post-LOCA core uncovery scenarios as a result of loop seal re-plugging do not constitute a significant concern to Seabrook Station plant safety.

Westinghouse proprietary information is provided in Enclosure 2.

References

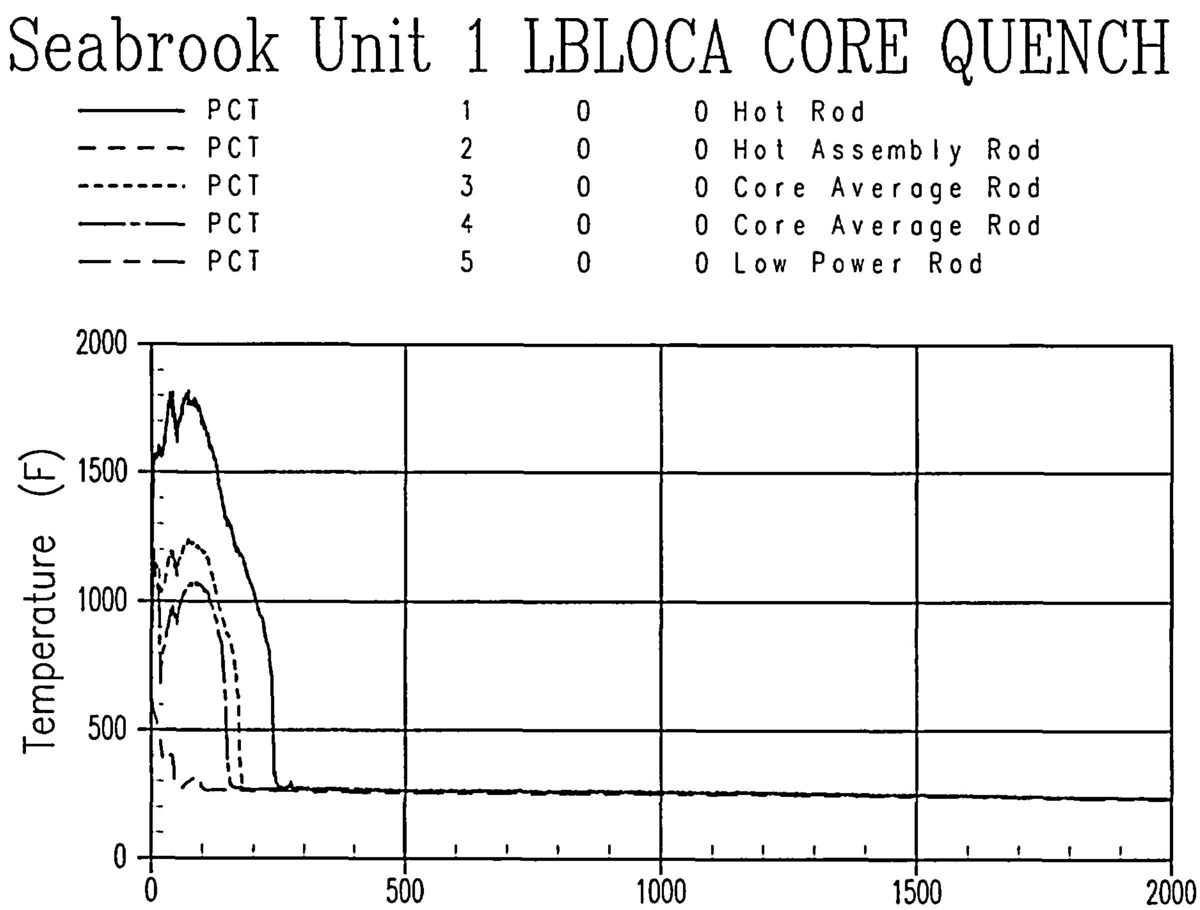
1. WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code", S. D. Rupprecht, et al., 1986.
2. WCAP-10054-P Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", C. M. Thompson, et al., July 1997.
3. OG-87-37, "Westinghouse Owners Group (WOG) Post LOCA Long Term Cooling, Letter from Roger Newton (WOG) to Thomas Murley (NRC)", August 26, 1987.
4. NSD-NRC-97-5092, "Core Uncovery Due to Loop Seal Re-Plugging During Post-LOCA Recovery," Letter from N. J. Liparulo (W) to NRC, March, 1997

RAI #37

Provide the LBLOCA analysis results via tables and graphs to the time that stable and sustained quench is established.

FPL Energy Seabrook Response:

In order to demonstrate stable and sustained quench, the WCOBRA/TRAC calculation for the maximum local oxidation analysis for Seabrook Station was extended. Figure RAI 37-1 shows the peak cladding temperatures for the five rods modeled in WCOBRA/TRAC. This figure indicates that quench occurs at approximately 100 seconds for the low power rod (rod 5), 175 seconds for the core average rods (rods 3 and 4), and around 250 seconds for the hot rod (rod 1) and hot assembly average rod (rod 2). Once quench is predicted to occur, the rod temperatures remain slightly above the fluid saturation temperature for the remainder of the simulation. Figure RAI 37-2 shows the collapsed liquid level in the four downcomer channels and shows steady behavior, with the level in each quadrant remaining near the bottom of the cold leg. By 1400 seconds, bulk boiling in the downcomer has been terminated, and subcooling in the downcomer has been re-established. Figure RAI 37-3 shows the collapsed liquid level in the four core channels and indicates a gradual increase in the core liquid inventory. This is consistent with the expected result based on the removal of the initial core stored energy and the gradual reduction in decay heat. Figure RAI 37-4 shows the vessel liquid mass and indicates an increasing trend beginning at about 300 seconds. This indicates that the increase in inventory due to the pumped safety injection is more than offsetting the loss of inventory through the break. Based on these results, it is concluded that stable and sustained quench has been established for the Seabrook Station Large Break LOCA analysis.



Time (Sec)

FIGURE RAI 37-1
Peak Cladding Temperatures

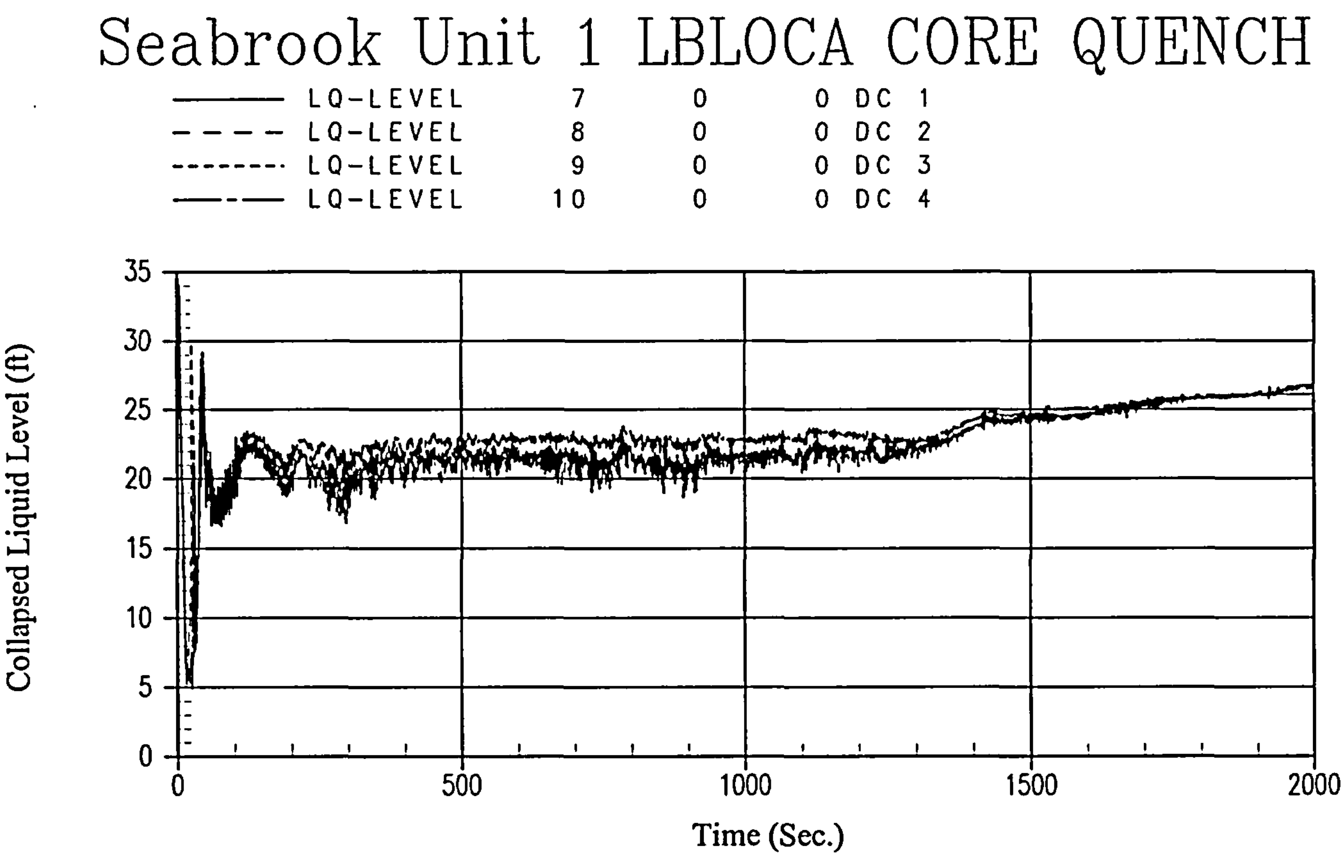


FIGURE RAI 37-2
Downcomer Collapsed Liquid Levels

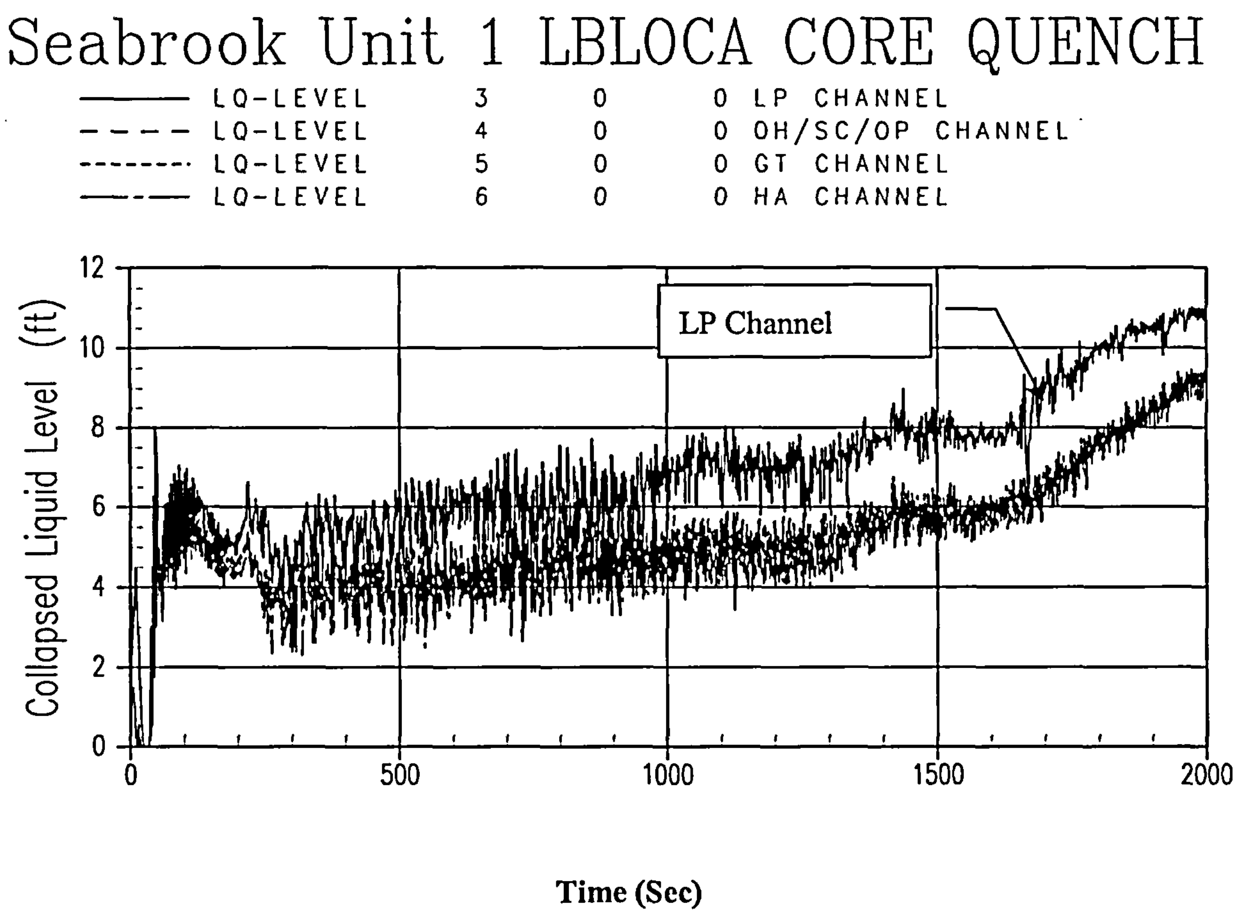
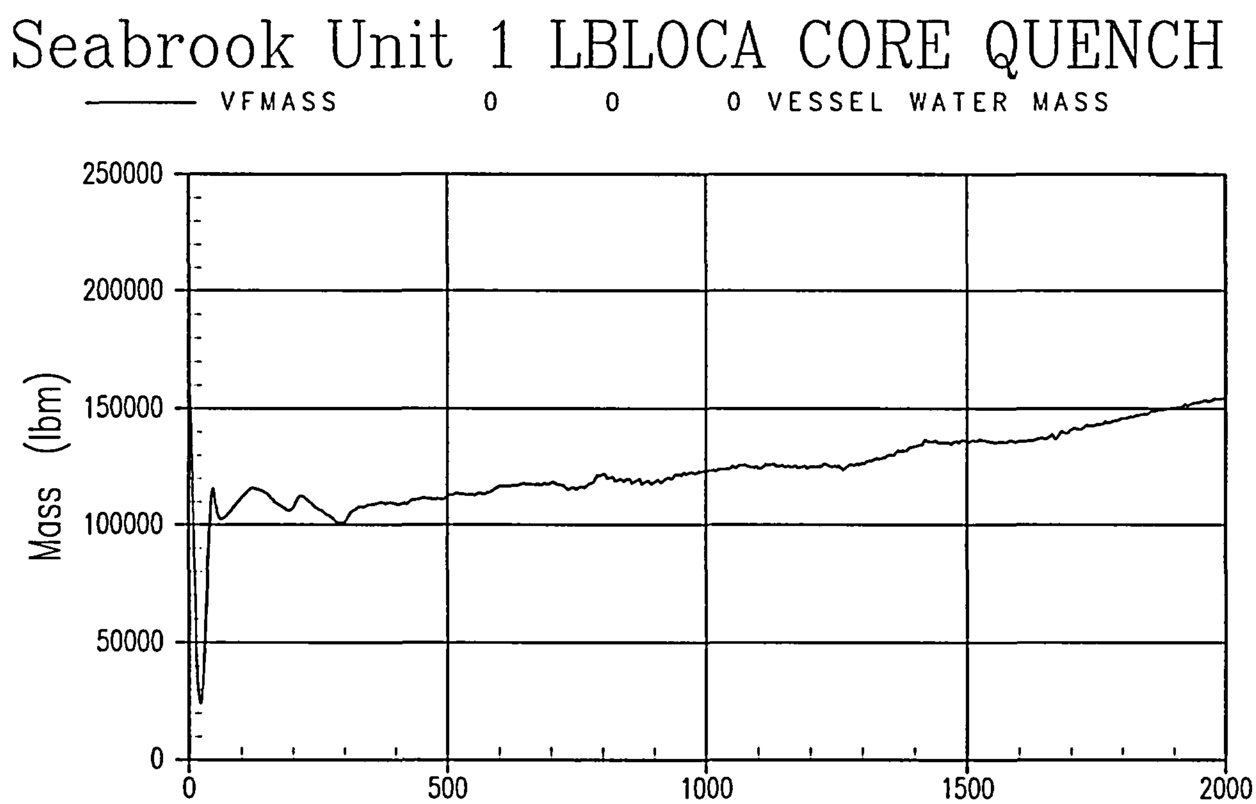


FIGURE RAI 37-3
Core Collapsed Liquid Levels



Time (Sec)

FIGURE RAI 37-4
Vessel Liquid Mass

RAI #38

It is not clear from LBLOCA and SBLOCA figures what specific upper core plate is used for Seabrook. Identify the specific upper core plate design used in Seabrook. Also identify whether Seabrook features a baffle/barrel upflow or baffle/barrel downflow design. Confirm that these features have been modeled in the input decks appropriately.

FPL Energy Seabrook Response:

The "inverted top hat" upper support plate design is modeled in the LBLOCA and SBLOCA analyses for the Seabrook Station SPU. The Seabrook Station barrel/baffle upflow design feature is also modeled in the LBLOCA and SBLOCA analyses for the SPU.

RAI #39

Tables 6.1.1-2 and 6.1.2.5-3 provide LBLOCA and SBLOCA analyses results for the Seabrook SPU.

Provide all results (peak clad temperature, maximum local oxidation, and total hydrogen generation) for both LBLOCA and SBLOCA. For maximum local oxidation, include consideration of both pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation. Also include the results for fuel resident from previous cycles.

FPL Energy Seabrook Response:

The results (peak cladding temperature, maximum local oxidation, and total hydrogen generation) for the Seabrook Station LBLOCA and SBLOCA SPU analyses are provided in Table RAI 39-1 below. Additional information regarding the bases for the maximum local oxidation, including consideration of both pre-existing and post-LOCA oxidation, cladding outside and post-rupture inside oxidation is discussed below.

LBLOCA

Per LAR Attachment 1, Table 6.1.1-2 (page 6-17), the transient maximum local oxidation calculated for the Seabrook Station SPU LBLOCA analysis is 3.53 percent. Consistent with the NRC-approved methodology, this value was calculated using a LOCA transient whose nominal peak cladding temperature exceeds the 95th percentile value for both the first and second reflood peaks. The transient maximum local oxidation was predicted to occur at the burst elevation, such that the metal-water reaction occurred on both the inner and outer cladding surfaces.

The maximum local oxidation was calculated for fresh fuel, at the beginning of the cycle. This represents the maximum amount of transient oxidation that could occur at any time in life. As burnup increases, the transient oxidation decreases for the following reasons:

1) The cladding creeps down towards the fuel pellets, due to the system pressure exceeding the rod internal pressure. This will reduce the average initial stored energy at the hot spot by several hundred degrees relatively early in the first cycle of operation. Accounting only for this change, which occurs early in the first cycle, reduces the transient oxidation significantly.

2) Later in life, the cladding creep-down benefit still remains in effect. In addition, with increasing irradiation, the power production from the fuel will naturally decrease as a result of depletion of the fissionable isotopes. Reductions in achievable peaking factors in the burned fuel relative to the fresh fuel are realized before the middle of the second cycle of operation. The achievable linear heat rates decrease steadily from this point until the fuel is discharged, at which point the transient oxidation will be negligible.

The pre-transient oxidation increases with burnup, from zero at beginning of life (BOL) to a maximum value at the discharge of the fuel (end of life, or EOL). The design limit 95% upper bound value for the SPU fuel design is <16%. The actual upper bound values predicted are expected to be well below this value.

Based on the above discussion, the transient oxidation decreases from a conservative maximum of 3.53% at beginning of life to a negligible value at end of life, while the pre-transient oxidation increases from zero at beginning of life to a conservative maximum at end of life of <16%. Additional WCOBRA/TRAC and HOTSPOT calculations were performed at an intermediate burnup, accounting for burnup effects on fuel performance data (primarily initial stored energy and rod internal pressure). These calculations support the conclusion that the sum of the transient and pre-transient oxidation remains below 16% at all times in life. This confirms Seabrook Station conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

SBLOCA

As part of the Seabrook Station SPU, a new SBLOCA analysis was performed. The break spectrum that was analyzed yielded a maximum peak clad temperature of 1373°F for a 4-inch equivalent break diameter. The break spectrum results are summarized in LAR Attachment 1 Tables 6.1.2.5-2 and 6.1.2.5-3 (pages 6-21 and 6-22). Because of low clad temperatures, fuel rod burst was not predicted to occur, and the maximum transient oxidation was only 0.20%.

The pre-transient oxidation increases with burnup, from zero at beginning of life (BOL) to a maximum value at the discharge of the fuel (end of life, or EOL). The design limit 95% upper bound value for the SPU fuel design is <16%. The actual upper bound values predicted are expected to be well below this value. Because the transient oxidation is so low, the sum of the transient and pre-transient oxidation remains below 16% at all times in life. This confirms Seabrook Station conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

TABLE RAI 39-1
SPU LOCA Analysis Results

	LBLOCA	SBLOCA
Peak Cladding Temperature	1784°F ($PCT^{95\%}$)	1373°F
Maximum Local Oxidation	Pre-transient = 0% Transient = <3.53%	Pre-transient = 0% Transient = 0.20%
Total Hydrogen Generation	0.3%	<<1%

RAI #40

Does the uprated power level or increased decay heat load affect the Emergency Core Cooling System (ECCS) switchover from injection mode to sump recirculation mode (timing in Emergency Operating Procedures (EOPs)) for Seabrook? Does this affect the Seabrook Updated Final Safety Analysis Report (UFSAR) Figure 6.3-10? Are ECCS pump net positive suction head analyses affected?

FPL Energy Seabrook Response:

There is no impact due to the SPU on the switchover time from injection mode to recirculation mode or on net positive suction head analyses for Emergency Core Cooling System pumps. Seabrook Station utilizes a semi-automatic switchover.

The switchover time for injection mode is based on the time it takes to drain the refueling water storage tank to the Lo-Lo level and the time required for all operator actions to complete pump suction switchover from the refueling water storage tank to the sump. The time to drain the refueling water storage tank is dependent upon refueling water storage tank available volume, instrument uncertainty, and maximum Emergency Core Cooling System flow rates during injection mode as dictated by system resistance and pump characteristics. None of these inputs are impacted by the SPU. Therefore, the SPU does not affect UFSAR Figure 6.3-10.

The net positive suction head requirements for the Emergency Core Cooling System pumps are dependent upon pump maximum flow rates, system resistance, and the elevation of the refueling water storage tank, sump, sump temperature, and Emergency Core Cooling System pump suction piping. The current net positive suction head analysis for the Emergency Core Cooling System pumps utilizes a very conservative basis per Regulatory Guide 1.1 (for example, 100°F fluid is assumed throughout the system and no credit is taken for containment overpressure). The current analysis bounds any effects of the SPU.

<u>RAI #41</u>

Provide the minimum time for switchover to hot leg injection and the basis for this time. Include, (a) the times specified in the EOPs that address switchover to hot leg injection, (b) a description of the applicable EOP (or a copy of the EOP), and (c) information that reasonably ensures the EOP actions will occur consistent with the stated times.

<u>FPL Energy Seabrook Response:</u>

The emergency operating procedure for switchover to hot leg injection includes a minimum and a maximum time to accomplish the switchover. The minimum time is based on ensuring there is adequate emergency core cooling flow for decay heat removal. This minimum time has been calculated to be 4 hours following the initiation of a loss of cooling accident (LOCA). The maximum time to accomplish the switchover to hot leg injection is based on precluding boron precipitation. As presented in LAR Attachment 1, Subsection 6.1.5.4 "Acceptance Criteria and Results" (page 6-12), this maximum time was calculated to be 7.46 hours following the initiation of a LOCA. Specific responses to the RAI are provided below.

a. The times specified in the emergency operating procedures that address switchover to hot leg injection will be a minimum time of 5.5 hours and a maximum time of 7 hours following the initiation of a LOCA.

b. There is a specific emergency operating procedure for hot leg injection entitled "Transfer to Hot Leg Recirculation". This emergency operating procedure is entered when the times specified in the loss of reactor or secondary coolant emergency operating procedure have elapsed, or when a decision is made, based upon the recommendation of the Technical Support Center, that transfer to hot leg injection is required. The times specified in the loss of reactor or secondary coolant emergency operating procedure for hot leg injection will be (1) prepare for hot leg injection 4 hours after event initiation, and (2) initiate hot leg injection no sooner than 5.5 hours and no later than 7.0 hours after event initiation. The minimum and maximum times will appear twice in the "Transfer to Hot Leg Recirculation" emergency operating procedure as cautions. The caution in both places will state, "The following actions should be performed no sooner than 5.5 hours and no later than 7.0 hours after cold leg injection begins".

c. Assurance is provided that hot leg injection emergency operating procedure actions will occur consistent with the calculated times because of two reasons: (1) there will be a margin of 1.5 hours (5.5 hours in the emergency operating procedures minus 4.0 hours calculated) for minimum time to hot leg injection, and a margin of 0.46 hours (7.46 hours calculated minus 7.0 hours in the emergency operating procedures) for maximum time to hot leg injection, and (2) based on discussions with Seabrook Station operations staff, the actions described in the hot leg injection emergency operating procedure can be performed in about 10 minutes.

RAI #42

Provide a copy of Reference 6.1-13, "Hot Leg Switchover Time Clarification," NSAL-04-01, January, 2004.

FPL Energy Seabrook Response:

Reference NSAL-04-1 is enclosed as Attachment RAI 42-1.

RAI #43

Table 6.3.1-4 summarizes the initial conditions and computer codes used which are approved by the NRC staff for non-LOCA transients analysis. For each computer code, the NRC staff provides a safety evaluation report which lists the staff's positions and limitations for its application. List the NRC staff approval status, the staff's positions or limitations for each computer code and address how Seabrook satisfies these requirements for SPU conditions.

FPL Energy Seabrook Response:

The computer codes used in the non-LOCA transient analyses, which are discussed in LAR Attachment 1, Subsection 6.3.1.9 "Computer Codes Utilized" (page 6-52), are listed below along with the associated topical report. Tables RAI 43-1 through RAI 43-6 provide the NRC approval status, safety evaluation report requirements, and justification for the Seabrook Station SPU.

- FACTRAN (WCAP-7908-A (Proprietary))
- RETRAN (WCAP-14882-P-A (Proprietary))
- LOFTRAN (WCAP-7907-P-A (Proprietary))
- Advanced Nodal Code (ANC) (WCAP-10965-P-A (Proprietary))
- TWINKLE (WCAP-7979-P-A (Proprietary))
- VIPRE (WCAP-14565-P-A (Proprietary))

Table RAI 43-1
FACTRAN
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for Seabrook

1. ***"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."***

   Justification

   The FACTRAN code was used in the analyses of the following transients for Seabrook: Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Condition (UFSAR 15.4.1) and RCCA Ejection (UFSAR 15.4.8). Conservative initial fuel temperatures were used as FACTRAN input in the RCCA Ejection analyses. The bounding fuel temperatures for these transients were calculated using the PAD 4.0 computer code (see WCAP-15063-P-A). As indicated in WCAP-15063-P-A, the method of determining uncertainties for PAD 4.0 fuel temperatures has been approved by the NRC.

2. ***"Table 2 presents the guidelines used to select initial temperatures."***

   Justification

   In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: Loss of Flow, Locked Rotor, and Rod Ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the RCCA Ejection analyses for Seabrook. The assumed fuel temperatures, which were based on bounding temperatures calculated using the PAD 4.0 computer code (see WCAP-15063-P-A), include uncertainties and are conservatively high.

3. ***"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."***

   Justification

   The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA Withdrawal from a Subcritical Condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize transient fuel temperatures. For the RCCA Ejection transients, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.

4. ***"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."***

   Justification

   Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for Seabrook (RCCA Withdrawal from a Subcritical Condition (UFSAR 15.4.1) and RCCA Ejection (UFSAR 15.4.8)).

Table RAI 43-1
FACTRAN
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

5.	***"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."*** Justification At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for Seabrook (RCCA Withdrawal from a Subcritical Condition (UFSAR 15.4.1) and RCCA Ejection (UFSAR 15.4.8)).
6.	***"Although time-independent mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-dependent mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."*** Justification The two transients that were analyzed with FACTRAN for Seabrook (RCCA Withdrawal from a Subcritical Condition (UFSAR 15.4.1) and RCCA Ejection (UFSAR 15.4.8)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA Withdrawal from a Subcritical Condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA Ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.
7.	***"The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."*** Justification The condition of concern (T(centerline) – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for Seabrook.

Table RAI 43-2
RETRAN
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

Computer Code:	RETRAN
Licensing Topical Report:	WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
Date of NRC Acceptance:	February 11, 1999 (SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for Seabrook

1. ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***

Justification

The transients listed in Table 1 of the SER are:

1 Feedwater system malfunctions,
2 Excessive increase in steam flow,
3 Inadvertent opening of a steam generator relief or safety valve,
4 Steam line break,
5 Loss of external load/turbine trip,
6 Loss of offsite power,
7 Loss of normal feedwater flow,
8 Feedwater line rupture,
9 Loss of forced reactor coolant flow,
10 Locked reactor coolant pump rotor/sheared shaft,
11 Control rod cluster withdrawal at power,
12 Dropped control rod cluster/dropped control bank,
13 Inadvertent increase in coolant inventory,
14 Inadvertent opening of a pressurizer relief or safety valve,
15 Steam generator tube rupture.

The transients analyzed for Seabrook using RETRAN are:

Excessive heat removal due to feedwater system malfunctions (UFSAR 15.1.2), (#1 above)
Steam line break (UFSAR 15.1.5) (#4 above)
Loss of external electrical load (UFSAR 15.2.2, 15.2.3), (#5 above)
Loss of AC power to the plant auxiliaries (UFSAR 15.2.6), (#6 above)
Loss of normal feedwater (UFSAR 15.2.7), (#7 above)
Feedwater system pipe break (UFSAR 15.2.8), (#8 above)
Loss of reactor coolant flow (UFSAR 15.3.1, 15.3.2), (#9 above)
Locked rotor / shaft break (UFSAR 15.3.3, 15.3.4), (#10 above)
Uncontrolled RCCA withdrawal at power (UFSAR 15.4.2), (#11 above)
Inadvertent operation of the ECCS at power (UFSAR 15.5.1) (#13 above)
CVCS malfunction that increases reactor coolant inventory (UFSAR 15.5.2) (#13 above)
Inadvertent opening of a pressurizer safety or relief valve (UFSAR 15.6.1) (#14 above)
Steam generator tube rupture (UFSAR 15.6.3) (#15 above)

As each transient analyzed for Seabrook using RETRAN is included in Table 1 of the WCAP-14882-P-A SER, no additional justification is required.

Table RAI 43-2
RETRAN
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

2.	***"WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."*** Justification Seabrook is a 4-loop Westinghouse-designed plant that was "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.
3.	***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."*** Justification The input data used in the RETRAN analyses performed by Westinghouse came from both Florida Power and Light Energy (FPL Energy) and Westinghouse sources. A quality assurance program is in place that required documentation of the input data sources and justification for use. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A, the safety analysis input values used in the Seabrook analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Table RAI 43-3
LOFTRAN
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

Computer Code:	LOFTRAN
Licensing Topical Report:	WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.
Date of NRC Acceptance:	July 29, 1983 (SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for Seabrook

1. ***"LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:***

    1- Feedwater System Malfunction
    2- Excessive Increase in Steam Flow
    3- Inadvertent Opening of a Steam Generator Relief or Safety Valve
    4- Steamline Break
    5- Loss of External Load
    6- Loss of Offsite Power
    7- Loss of Normal Feedwater
    8- Feedwater Line Rupture
    9- Loss of Forced Reactor Coolant Flow
    10- Locked Pump Rotor
    11- Rod Withdrawal at Power
    12- Rod Drop
    13- Startup of an Inactive Pump
    14- Inadvertent ECCS Actuation
    15- Inadvertent Opening of a Pressurizer Relief or Safety Valve

    This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture..."

    Justification

    For the Seabrook SPU, the LOFTRAN code was only used in the analyses of the rod drop transient (UFSAR 15.4.3) and ATWS (UFSAR 15.8). Rod drop is included in the list of transients presented in the SER.

    Although ATWS is not included in the SER list of transients presented above, Section 6 of the SER states that "LOFTRAN has also been reviewed by the staff for application to the Anticipated Transients Without Scram (ATWS) issue," and the "acceptability of LOFTRAN...has been accounted for in this review."

Table RAI 43-4
Advanced Nodal Code
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

Computer Code:	Advanced Nodal Code
Licensing Topical Report:	WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.
Date of NRC Acceptance:	June 23, 1986 (SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for Seabrook	
There are no conditions, restrictions, or limitations cited in the Advanced Nodal Code safety evaluation report..	

Table RAI 43-5
TWINKLE
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

Computer Code:	TWINKLE
Licensing Topical Report:	WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code," January 1975.
Date of NRC Acceptance:	July 29, 1974 (SER from D. B. Vassallo (U.S. Atomic Energy Commission) to R. Salvatori (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for Seabrook	
There are no conditions, restrictions, or limitations cited in the Advanced Nodal Code safety evaluation report..	

Table RAI 43-6
VIPRE
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

Computer Code:	VIPRE
Licensing Topical Report:	WCAP-14565-P-A, VIPRE-01 *Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Y. Sung, et al., October 1999.
Date of NRC Acceptance:	Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14565, 'VIPRE-01 *Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis,' (TAC No. M98666)," January 19, 1999.*

Safety Evaluation Report (SER) Conditions & Justification for Seabrook

1. ***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."***

   Justification

   The NRC-approved WRB-2M correlation was used in the DNBR analyses. Justification of the WRB-2M correlation limit of 1.14 with the VIPRE code is provided in WCAP-14565-P-A.

   For the Seabrook SPU DNBR analyses, the plant specific hot channel factors for enthalpy rise and other fuel-dependent parameters that have been previously approved by the NRC have been assumed in these analyses.

2. ***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."***

   Justification

   The core boundary conditions for the VIPRE calculations are all generated from NRC-approved methodologies and computer codes, such as RETRAN and ANC. Conservative reactor core boundary conditions were justified for use as input to VIPRE as discussed in the safety evaluations. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272/9273.

3. ***"The NRC Staff's generic SER for VIPRE (Reference 2 of the SER) set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."***

   Justification

   Justification on use of the WRB-2M correlation with the VIPRE code is provided in WCAP-14565-P-A.

Table RAI 43-6
VIPRE
Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes

4. ***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE (Reference 2 of the SER) did not extent to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."***

Justification

Justification

For the Seabrook SPU analyses, the use of VIPRE in the post-CHF region is limited to the peak clad temperature calculations for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P-A and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient;
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation;
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Conservative results were further ensured with the following inputs:

- Fuel rod input based on the maximum fuel temperature at the given power;
- The hot spot power factor was equal to or greater than the design linear heat rate;

Uncertainties were applied to the initial operating conditions in the limiting direction.

RAI #44

Provide the safety injection flowrate, as a function of pressure, that is assumed for the steam line break and other non-LOCA analyses. For the steam line break analysis, identify the most restrictive active single failure postulated for the safety injection system.

FPL Energy Seabrook Response:

For events where minimum Emergency Core Cooling System flow is conservative (Steam Line Break (UFSAR 15.1.5) and Feedwater System Pipe Break (UFSAR 15.2.8)), the following curve (Table RAI 44-1), which reflects the most restrictive single active failure of one safety injection pump train, is assumed.

TABLE RAI 44-1

RCS Pressure (psia)	SI Flow (lbm/sec)	RCS Pressure (psia)	SI Flow (lbm/sec)
15	54.97	1615	31.94
215	52.73	1815	28.11
415	50.28	2015	23.37
615	47.66	2215	16.36
815	44.92	2315	9.40
1015	41.88	2355	0.00
1215	38.85	5000	0.00
1415	35.49	-----	-----

For events where maximum Emergency Core Cooling System flow is conservative (Inadvertent Operation of the Emergency Core Cooling System at Power (UFSAR 15.5.1)), the following curve (Table RAI 44-2) is assumed.

TABLE RAI 44-2

RCS Pressure (psia)	SI Flow (lbm/sec)	RCS Pressure (psia)	SI Flow (lbm/sec)
1615	70.53	2415	36.40
1815	64.83	2515	29.10
2015	56.95	3000	0.00
2215	48.49	5000	0.00

RAI #45

Provide the results of thermal hydraulic analysis of a SG tube rupture for radiological consequences including sequence of events, major assumptions, and transient curves.

FPL Energy Seabrook Response:

By letter NYN-03-061 dated October 6, 2003, FPL Energy Seabrook, LLC submitted License Amendment Request (LAR) 03-02 "Alternate Source Term." The analyses to support this LAR were performed at the analyzed SPU core power level of 3659 megawatts thermal discussed in LAR 04-03. For the steam generator tube rupture event, the thermal hydraulic analysis for radiological consequences, including sequence of events, margin assumptions, and transient curves is presented in LAR 03-02, which is currently under review by the NRC.

RAI #46

Confirm that the thermal hydraulic analysis of a SG tube rupture for radiological consequences, is performed with the most limiting single failure of a stuck open atmospheric steam dump valve associated with the failed SG and a concurrent loss of off-site power. Explain why the failure of an intact SG atmospheric steam dump valve to open during cooldown is assumed to be limiting with respect to margin to SG overfill.

FPL Energy Seabrook Response:

See FPL Energy Seabrook response to RAI #45. The thermal hydraulic analysis of the steam generator tube rupture event for radiological consequences was performed with the most limiting single failure of a stuck open steam generator atmospheric steam dump valve associated with the faulted steam generator and a concurrent loss of offsite power. (See page 30 of Enclosure 2 to FPL Energy Seabrook letter NYN-03-061.)

With regard to the SPU margin to steam generator overfill analysis, single failures were evaluated to determine the most limiting failure. Note that the most limiting single failure identified in the current Seabrook Station margin to steam generator overfill analysis is failure of a steam generator atmospheric steam dump valve on one of the intact steam generators (see Seabrook Station UFSAR, Subsection 15.6.3.2.a, page 9).

Single failures evaluated for the SPU were failures of the Emergency Feedwater System and steam generator atmospheric steam dump valves on the intact steam generators. If the single failure considered was a failure of the ruptured steam generator atmospheric steam dump valve to close after the initial lift, the flow out of the failed valve will exceed break flow into the faulted steam generator minimizing the effect on steam generator overfill. The failure of an atmospheric steam dump valve to open on an intact steam generator results in a longer time to cooldown the Reactor Coolant System, which delays depressurization of the Reactor Coolant System. This delays the termination of safety injection, delays break flow termination, and leads to less margin to overfill than a single failure of the steam generator atmospheric steam dump valve on the faulted steam generator or Emergency Feedwater System. Specific single failures were analyzed in the NRC approved methodology described in WCAP-10698-P-A.

The SPU analysis results demonstrate that a failure of a steam generator atmospheric steam dump valve on the intact steam generator to open on demand at the initiation of the Reactor Coolant System cooldown is the limiting single failure leading to the smallest margin to overfill. This failure is consistent with the limiting failure currently in the Seabrook Station UFSAR.

Operator actions and times required to stabilize the plant and terminate primary to secondary leakage are described in LAR Attachment 1 Subsection 6.2.2.2 (page 6-35). The operator actions and times analyzed for the SPU are consistent with the current licensing basis analysis and are specifically identified in emergency operating procedures. Time studies have been performed to demonstrate the ability of operators to perform the required actions within the analyzed time frames.

RAI #47

In its letter of January 17, 1989, "Acceptance for Referencing Topical Report WCAP-11397, 'Revised Thermal Design Procedure'," the NRC staff stated that, "Sensitivity factors used for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal." Supply the sensitivity factors, and their ranges of applicability for Seabrook.

FPL Energy Seabrook Response:

Sensitivities vary with the Revised Thermal Design Procedure (RTDP) conditions analyzed and cell type (typical or thimble). For each Revised Thermal Design Procedure parameter, the ranges and sensitivities for the SPU are given in Table RAI 47-1 below.

TABLE RAI 47-1
REVISED THERMAL DESIGN PROCEDURE
RANGES AND SENSITIVITIES

Parameter	Range for DNBR Analyses	DNBR Sensitivities [a]
Pressure (psia)	1775-2425	0.00 to 1.55
Power (%)	71.3-125	-1.34 to -2.33
Flow (%)	60-100	0.61 to 1.49
T_{in}, (°F)	554.5-644.0	-2.80 to -8.21
$F_{\Delta H}$	1.587-2.287	-0.56 to -2.73
$F^E_{\Delta H,1}$	1.000-1.021	0.00 to -0.17

Note:

(a) The sensitivities are in terms of ∂(DNBR) / ∂(Parameter), (% / %). For example, the percent change in DNBR per percent change in pressure has a sensitivity range of 0.00 to 1.55.

RAI #48

Discuss the implications of a failure to select the correct reference T_{AVG} in the overtemperature ΔT and overpower ΔT setpoint equations (i.e., to select a reference T_{AVG} that corresponds to the desired operating T_{AVG} in the T_{AVG} window), upon the accident analyses for events that rely upon the overtemperature ΔT and overpower ΔT reactor trips for protection.

FPL Energy Seabrook Response:

The use of a higher Overtemperature ΔT / Overpower ΔT reference T_{avg} than the reference full power T_{avg} has the effect of moving the Overtemperature ΔT and the Overpower ΔT setpoints in a non-conservative direction by the ratio of the ΔT corresponding to the reference T_{avg} used in the Overtemperature ΔT / Overpower ΔT setpoints to the actual ΔT corresponding to the selected full power T_{avg}. With the plant operating at the bottom of the T_{avg} window and the Overtemperature ΔT / Overpower ΔT reference T_{avg} set to the high end of the window, the Overtemperature ΔT / Overpower ΔT setpoints will be non-conservative by less than 3% of nominal full power ΔT. Based on the calculated margin to applicable acceptance criteria, the analyses of the transients for which a ΔT trip is credited for an incorrect reference temperature would be shown to be acceptable.

RAI #49

Please itemize the delays making up the 77-second delay to start the emergency feedwater pump.

FPL Energy Seabrook Response:

The Emergency Feedwater System has two redundant 100% capacity pumps. One is motor-driven and the other is steam turbine-driven. Since the delay to start the turbine-driven emergency feedwater pump is greater than that of the motor-driven emergency feedwater pump, the delay for the turbine-driven pump is used in the accident analyses. The starting sequence for the turbine-driven pump includes: the Emergency Safety Features Actuation Signal response time, opening time for the steam supply valves, a time delay to allow for condensate accumulated in the steam supply lines to be swept clear, opening time for the turbine steam admission valve, and the time for the pump to develop the speed required to provide design head and flow.

The time sequencing for initiation of Emergency Feedwater System flow is provided in Table RAI 49-1 below:

TABLE RAI 49-1

Emergency Feedwater System Flow Starting Sequence

EVENT	TIME (seconds)
Emergency safety features response time	2
Open steam supply valves	12
Time delay to clear condensate from steam line	28
Open pump turbine steam admission valve	15
Emergency feedwater pump turbine develop speed and flow	8
Design margin	12

RAI #50

Has the RCP coastdown flow predicted by RETRAN for the accident analyses presented in the March 17, 2004, submittal been validated against plant data? If not, justify its validity.

FPL Energy Seabrook Response:

The reactor coolant pump coastdown flow predicted by RETRAN for the Seabrook Station SPU analyses has not been validated against plant-specific data. The original Seabrook Station startup flow coastdown test was performed in the late 1980s, and it demonstrated that the rate of decrease in actual flow was not faster than that calculated for the Seabrook Station UFSAR analyses, which were performed using the LOFTRAN computer code. As RETRAN code results were shown to compare favorably to LOFTRAN code results in WCAP-14882-P-A, RETRAN is considered equivalent to LOFTRAN. Historically, the LOFTRAN code has been compared to a large variety of plant operational and test transients for 2-, 3-, and 4-loop plants. LOFTRAN best estimate predictions have been used in the area of flow coastdown, turbine trip, and natural circulation tests to support plant startup tests. In addition, LOFTRAN is extensively used in performing operational transient calculations to assist in plant control operability. Based on this extensive amount of verification and the favorable comparison between RETRAN results and LOFTRAN results shown in WCAP-14882-P-A, additional plant data comparisons are not necessary.

RAI #51

In Table 1 of WCAP-8567-P-A, "Improved Thermal Design Procedure," upon which the Revised Thermal Design Procedure (RTDP) is based, is limited to $F_{\Delta H}$ factors of 1.61 or less. Tables 6.1.2-1 and 7.2-1 simply indicate that the $F_{\Delta H}$ is 1.65, whereas Tables 6.3.1-2 and Table 7.1-1 indicate that a "statistical" $F_{\Delta H}$ of 1.587 is used. Is the latter value of $F_{\Delta H}$ used only with the RTDP? If not, then justify the use of an $F_{\Delta H}$ of 1.587 with standard thermal design procedures.

FPL Energy Seabrook Response:

As stated in the title of Table 1 of WCAP-8567-P-A, the $F_{\Delta H}$ values are represented as "typical" values. These should not be considered as limit values. The $F_{\Delta H}$ values used for the Seabrook Station SPU Revised Thermal Design Procedure (RTDP) analysis are Seabrook specific. The Standard Thermal Design Procedures (STDP) $F_{\Delta H}$ value is 1.65 and the Revised Thermal Design Procedure (RTDP) value is 1.587. Appropriate design $F_{\Delta H}$ values were used in the analysis of each accident.

RAI #52

Table 6.1.2-1 indicates that the vessel average temperature uncertainty is +3/-6°F (represents a ±3°F uncertainty plus a -3°F bias); but temperature uncertainty is defined elsewhere (Sections 6.1.7.2 and 6.4.1.1.1) as ±6°F. A ±3°F uncertainty plus a -3°F bias would yield 0/-6°F, and a ±6°F uncertainty would yield +3/-9°F. Which is correct? What is the source of the bias?

FPL Energy Seabrook Response:

As shown in LAR Attachment 1, Table 4.3-1 (page 4-31), the final, calculated, initial condition uncertainties include ±2.9°F random uncertainty and a -2.8°F bias. For analysis purposes, these values were conservatively rounded to a minimum of ±3.0°F random and -3.0°F bias. For simplicity, bounding values for the random uncertainty and bias were combined and conservatively applied in the safety analyses, e.g., ±6°F or +6/-5°F. The following two bullets explain the relationship between instrument uncertainty/bias values and actual (initial) condition values.

- With a positive instrument uncertainty/bias, the instrument channel indicates higher than the actual temperature (actual temperature is lower than the nominal value).
- With a negative instrument uncertainty/bias, the instrument channel indicates lower than the actual temperature (actual temperature is higher than the nominal value).

As the bias is only valid in one direction, a ±3.0°F random instrument channel uncertainty plus a -3.0°F bias will yield maximum offsets of +3.0°F and -6.0°F. This translates into a maximum actual initial condition T_{avg} that is 6.0°F above the nominal value and a minimum actual initial condition T_{avg} that is 3.0°F below the nominal value.

RAI #53

Notes (1) and (2) of Table 6.3.1-4 state that the initial average temperatures, assumed for certain accident analyses, were 577°F and 584.1°F, and that these temperatures were determined from sensitivity studies. Describe the sensitivity studies and explain the phenomena that led to these conclusions.

FPL Energy Seabrook Response:

LAR Attachment 1 Table 6.3.1-4 (page 6-65) Note 2, which indicates 577°F (low T_{avg} plus uncertainties) was found to be limiting for Loss of Normal Feedwater/Loss of AC Power, is not valid because high T_{avg} minus uncertainties was found to be limiting for these transients. LAR Attachment 1, Table 6.3.1-4 is revised to delete Note 2 for Loss of Normal Feedwater/Loss of AC Power transient. That leaves only Note 1 as being applicable to the Loss of Normal Feedwater / Loss of AC Power to the Plant Auxiliaries, Inadvertent Operation of Emergency Core Cooling System during Power Operation, and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory transients. Note 1 indicates that 584.1°F (high T_{avg} minus uncertainties) produces the limiting results for these transients, based on transient-specific sensitivity studies.

The sensitivity studies simply involved analyzing a matrix of cases that differ by the assumed initial T_{avg} value (high T_{avg} plus/minus uncertainties and low T_{avg} plus/minus uncertainties). Related to the initial T_{avg} is the initial pressurizer level, which is based on either the high T_{avg} program or the low T_{avg} program. Whereas the high T_{avg} (589.1°F) corresponds to a nominal pressurizer level of 60% span, the low T_{avg} (571.0°F) corresponds to a nominal pressurizer level of 33.1% span; an additional +5% span is applied in each case to account for initial condition uncertainties. Both the initial T_{avg} and the initial pressurizer level are critical parameters for the Loss of Normal Feedwater / Loss of AC Power to the Plant Auxiliaries, Inadvertent Emergency Core Cooling System, and Chemical and Volume Control System Malfunction transients, which are analyzed mainly for pressurizer filling concerns. Typically, a high initial pressurizer level corresponding to high T_{avg} is conservative because it minimizes the initial margin to filling the pressurizer. However, a lower initial pressurizer level may be more limiting because the reactor coolant will expand more starting from the corresponding low T_{avg}. There are other factors such as emergency feedwater capacity and control systems actuations that can influence which initial conditions are limiting. A matrix of cases is examined to determine the limiting initial conditions. For Seabrook Station, the sensitivities demonstrated that it is more conservative to minimize the initial margin to pressurizer fill by assuming the pressurizer level corresponding to the high T_{avg}. Also, the high T_{avg} minus uncertainty (584.1°F) was found to be the most limiting T_{avg} assumption for the Loss of Normal Feedwater / Loss of AC Power to the Plant Auxiliaries, Inadvertent Emergency Core Cooling System, and Chemical and Volume Control System Malfunction transients.

Finally, reference to Notes (1) and (2) incorrectly appears in the "DNB Correlation" column for the Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition transient. LAR Attachment 1, Table 6.3.1-4 for the Uncontrolled Rod Cluster Assembly from Subcritical Condition transient is revised to delete Notes 1 and 3 and to add Note 3. Note 3 is added at the bottom of the table stating, "The W-3 correlation was used for departure from nucleate boiling ratio calculations below the first mixing vane grid, and the WRB-2 correlation was used for departure from nucleate boiling ratio calculations above the first mixing vane grid."

RAI #54

The NRC request for additional information did not include an RAI #54.

RAI #55

In Section 6.3.1.2, "Overtemperature ΔT and Overpower ΔT Reactor Trip Setpoints," FPLE states the following:

> The revised safety analysis setpoints are based upon the assumption that the reference average temperature (T') used in the overtemperature ΔT and overpower ΔT setpoint equations correspond to the selected operating temperature within the T_{AVG} window.

How can one be sure that the correct T' is being used in the overtemperature ΔT and overpower ΔT setpoint equations, at any given time during operations? How will the T' be changed whenever the operating temperature is changed (within the T_{AVG} window)?

FPL Energy Seabrook Response:

At the beginning of each refueling cycle, a design (or reference) T_{avg} is chosen for that fuel cycle as part of the reload safety analysis checklist process. The chosen T_{avg} must be within the analyzed T_{avg} window. Once chosen, the design T_{avg} is fixed for that operating cycle (i.e., the plant does not change design T_{avg} during the cycle). The above quoted safety analysis assumption is that the plant will operate at the chosen design T_{avg}, and the ΔT trip settings for T' and T" will be at the design T_{avg}. However, during the course of operation, indicated full power T_{avg} may change due to changes in streaming characteristics as a result of core burndown and may not be the same for each loop. To account for a change in indicated T_{avg} due to streaming, and to account for human factors ease in setting the ΔT trip setpoints to a single value for all the loops, an allowance of 3.8°F is allocated in the safety analysis for indicated T_{avg} deviation from the setting for T' and T" in the ΔT trip setpoints. This criterion is verified on a quarterly basis via plant surveillance procedures.

RAI #56

In Section 6.3.2.1.3, "Results," FPLE states the following:

> The reduction in feedwater temperature due to a 10-percent step load increase is greater than 35°F. The increased thermal load, due to the opening of the low-pressure heater bypass valve, thus results in a transient very similar, but of reduced magnitude, to the steam system piping failure initiated from full power conditions described in [License Amendment Request] LAR Section 6.3.2.4. No transient results are presented, as no explicit analysis is performed.

Similarly, in Section 6.3.2.1.4, "Conclusions," FPLE states:

> With respect to the feedwater temperature reduction transient (accidental opening of the feedwater bypass valve), it was determined to be less severe than the steam system piping failure initiated at full power conditions (see Seabrook Station UFSAR Section 15.1.5); no explicit analysis is performed.

Section 6.3.2.4, "Steam System Piping Failure," and Seabrook UFSAR Section 15.1.5 describe the analysis of the steam system piping failure initiated from zero power conditions, only. How is the reduction in feedwater temperature, a Condition II event analyzed at hot full power (HFP), comparable to the steam system piping failure, a Condition IV event analyzed at hot full power (HZP)? Typically, the reduction in feedwater temperature event is compared to the excessive load increase event.

FPL Energy Seabrook Response:

A steam flow increase of slightly more than 5% was determined to be equivalent to the maximum expected feedwater temperature reduction of 35°F. As the excessive load increase event corresponds to a steam flow increase of 10%, it bounds the feedwater temperature reduction event.

RAI #57

In Section 6.3.2.3.1, "Accident Desciption," FPLE states that:

> .... a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves and trip the main feedwater pumps.

Tables 6.3.2.1-1 and 6.3.2.1-2 do not show the generation of a safety injection signal. Instead, they list the time the hi-hi steam generator water level trip setpoint is reached, and the time the turbine is tripped (two seconds later). Both tables indicate that the feedwater isolation valves are fully closed ten seconds after the turbine is tripped and 12 seconds after the hi-hi SG water level trip setpoint is reached. Shouldn't the feedwater isolation valves be fully closed 12 seconds after the turbine is tripped (i.e., the turbine trip/feedwater isolation signal is generated)?

FPL Energy Seabrook Response:

LAR Attachment 1, Subsection 6.3.2.3.1 "Accident Description" (page 6-78) contains the Accident Description for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event, which is bounded by the Steam System Piping Failure event discussed in LAR Attachment 1, Subsection 6.3.2.4 "Steam System Piping Failure" (page 6-79). LAR Attachment 1, Table 6.3.2.4-1 (page 6-85) is related to the Steam System Piping Failure event, and it does show the generation of a safety injection signal. LAR Attachment 1, Tables 6.3.2.1-1 and 6.3.2.1-2 (page 6-84) correspond to the full power and zero power Feedwater System Malfunction events, respectively, which do not result in the generation of a safety injection signal.

As for the question on feedwater isolation, Table 3.3-3 of the Seabrook Station Technical Specifications shows that feedwater isolation is actuated off of either a high-high steam generator water level signal or a safety injection signal, but not a turbine trip.

RAI #58

In Section 6.3.3.3.2, why are the RCPs assumed to lose power and begin coasting down 2 seconds after the reactor trip on low-low SG water level, and not at the same time as the reactor trip signal is received?

FPL Energy Seabrook Response:

The Loss of Offsite Power (LOOP) event is analyzed to demonstrate the long-term heat removal capability of the Emergency Feedwater System. For this, the effect that the Loss of Offsite Power (at T = 0 seconds) has on reactor coolant pump operation is conservatively delayed until after the reactor trip occurs on low-low steam generator level to allow a maximum depletion of the steam generator inventory. This loss of secondary inventory provides a severe limitation on the long term cooling capability of the secondary in comparison to a case where the Loss of Offsite Power would result in reactor coolant pump coastdown at time zero when steam generator levels are near normal operating level. For this specific transient the reactor coolant pump coastdown delay time after reactor trip is not an important or critical parameter.

RAI #59

Explain the drop in pressurizer pressure, occurring at about 10 minutes in Figure 6.3.3.3-3.

FPL Energy Seabrook Response:

The apparent drop in pressurizer pressure shown on the figure at ~600 seconds is not a real phenomenon, but is an anomalous result of the time step selected in the RETRAN input. This same case was rerun with a slightly smaller maximum time step to evaluate the sensitivity of the results to the time step chosen, and the pressure drop in question did not occur. See Figure RAI 59-1. The pressurizer water volumes for the two cases were compared, and the case presented in the LAR Attachment 1 Subsection 6.3.3.3 (page 6-109) (base case) results in a slightly higher (more conservative) long-term peak value, although the difference is not significant (1463 ft^3 versus 1456 ft^3). See Figure RAI 59-2. Therefore, although the pressure drop shown in LAR Attachment 1, Figure 6.3.3.3-3 (page 6-146) is not a real phenomenon, it does not have a significant impact on the results of the analysis. Figures RAI 59-3 and RAI 59-4 present the pressurizer pressure and pressurizer water volume plots, respectively, for the slightly less limiting time step sensitivity case.

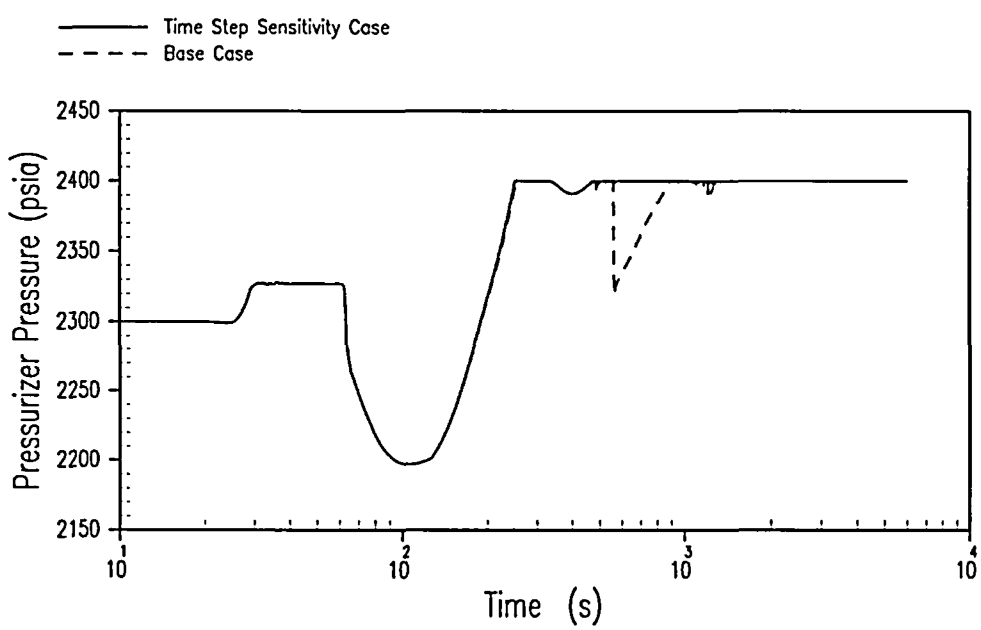


FIGURE RAI 59-1
LOSS OF NONEMERGENCY AC POWER TO THE PLANT AUXILIARIES
PRESSURIZER PRESSURE – CASE COMPARISON

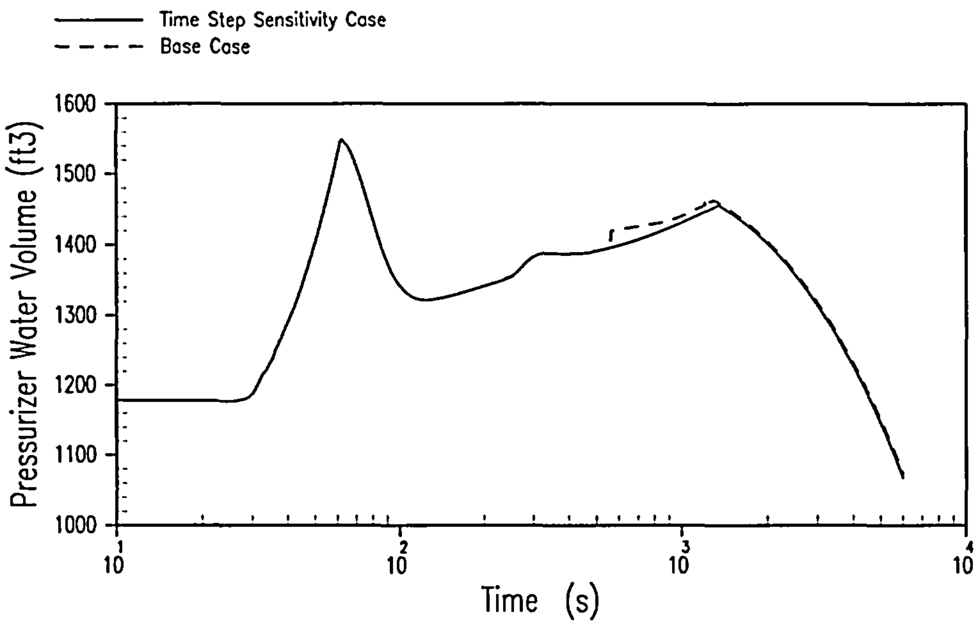


FIGURE RAI 59-2
LOSS OF NONEMERGENCY AC POWER TO THE PLANT AUXILIARIES
PRESSURIZER WATER VOLUME – CASE COMPARISON

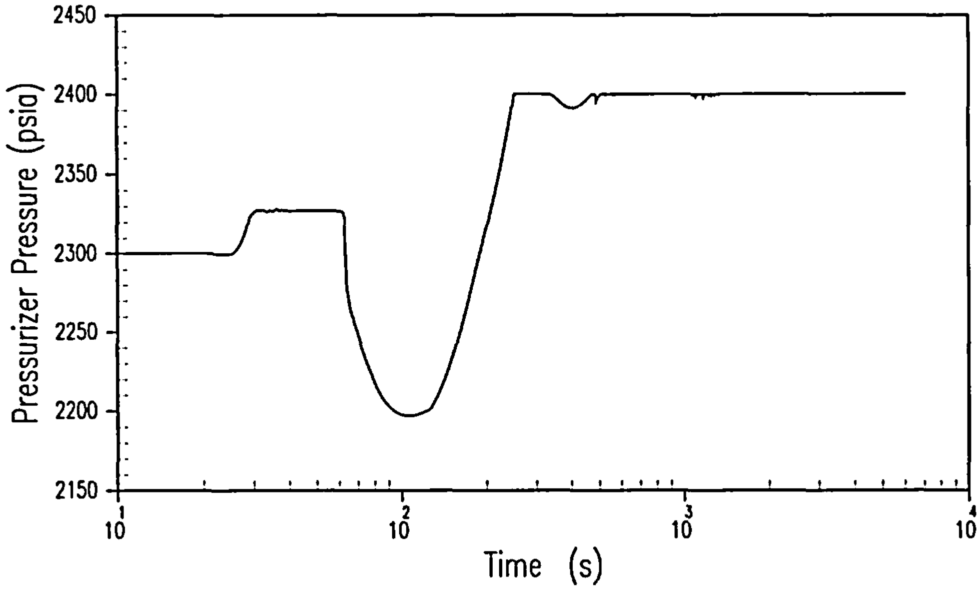


FIGURE RAI 59-3
LOSS OF NONEMERGENCY AC POWER TO THE PLANT AUXILIARIES
PRESSURIZER PRESSURE – REVISED TIME STEP

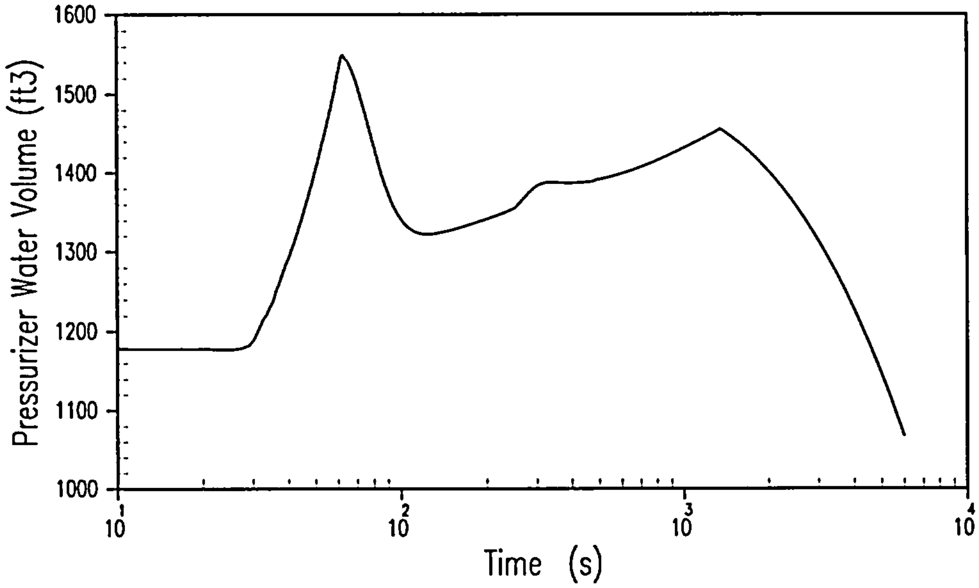


FIGURE RAI 59-4
LOSS OF NONEMERGENCY AC POWER TO THE PLANT AUXILIARIES
PRESSURIZER WATER VOLUME – REVISED TIME STEP

RAI #60

According to Figures 6.3.3.3-3 and 6.3.3.3-4, the pressurizer relief valves open, after about 250 seconds, and relieve steam. Identify the single applicable acceptance criterion, indicated in Section 6.3.3.3.4, that is satisfied, and confirm that no RCS water is relieved from the pressurizer.

FPL Energy Seabrook Response:

The single applicable acceptance criterion for the Loss of Offsite Power is the same as for the Loss of Normal Feedwater, i.e., the pressurizer does not become water solid. As noted in LAR Attachment 1, Table 6.3.3.3-1 (page 6-120) and Figure 6.3.3.3-4 (page 6-147), the long-term peak water volume in the pressurizer is 1463 ft^3 versus the limit of 1834.4 ft^3. Therefore, the criterion is met and Reactor Coolant System water is not relieved from the pressurizer.

RAI #61

In Section 6.3.2.4.1, "Accident Description," FPLE states that:

> Following a steam line break, the core is ultimately shut down by the boric acid injected into the Reactor Coolant System by the Safety Injection System.

Although Figure 6.3.2.4-11 indicates that the core boron concentration continues to increase as safety injection fluid is added by the safety injection pumps and supplemented by the accumulators, the core reactivity, in Figure 6.3.2.4-1, does not become subcritical. The safety injection pumps reach full speed at about 28 seconds (Table 6.3.2.4-1); but the heat flux doesn't peak, and the minimum departure from nucleate boiling ratio is not reached until about two minutes later. Please explain this.

FPL Energy Seabrook Response:

Although it is difficult to see in LAR Attachment 1, Figure 6.3.2.4-1 (page 6-90) due to the scale used to show the full transient, the core reactivity does become slightly negative at ~223 seconds and remains negative for the duration of the transient. The reason why it doesn't become more negative is because there is a relatively equal balance between the positive reactivity insertion caused by the continued cooldown of the Reactor Coolant System (due to the steam release from the break and the cold safety injection water) and the negative reactivity from the injected boron.

The delay between when the safety injection pumps reach full speed and the time of the peak heat flux/minimum departure from nucleate boiling ratio is attributed to the assumption that the safety injection lines down stream of the refueling water storage tank initially contain unborated water. This assumption conservatively maximizes the time it takes to deliver the highly concentrated refueling water storage tank boric acid solution to the reactor coolant loops.

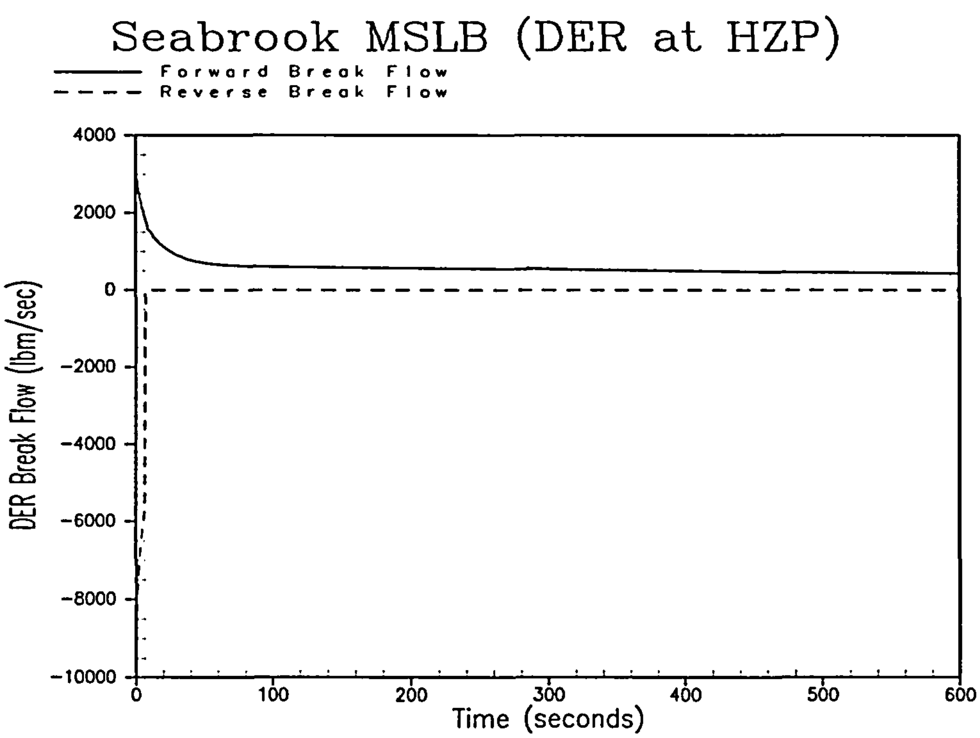
RAI #62

Explain why steam flow in the faulted loop (Figure 6.3.2.4-7), goes to zero at less than ten seconds into the transient. Did the analysis assume that the main steam isolation valve (MSIV) in the faulted loop would fail to close?

FPL Energy Seabrook Response:

The two plots in LAR Attachment 1, Figure 6.3.2.4-7 (page 6-96) reflect the steam flow through the faulted loop's main steam isolation valve and the steam flow through the main steam isolation valve of one of the intact loops. The faulted loop main steam isolation valve flow is negative because it represents the steam flow from the intact loops and the header. The main steam isolation valve closure that occurs 6 seconds after the low steam line pressure setpoint is reached isolates the intact loops and steam header from the break, thus leaving only the faulted loop blowing down. Figure RAI-62-1 provides a plot of the forward and reverse break flows. A failure of the faulted loop main steam isolation valve was not assumed, however, such a failure would only allow the steam header to continue to blow down, which does not affect the Reactor Coolant System transient.

FIGURE RAI 62-1
Seabrook MSLB DER Forward and Reverse Break Flows



RAI #63

In Table 6.3.1-4, the initial average temperatures assumed for the loss of normal feedwater accident analysis is a value of 566°F. What is the basis for this temperature?

FPL Energy Seabrook Response:

Sensitivity analyses were performed at the high and low end of the reactor coolant average temperature range (589.1°F and 571.0°F), with plus and minus uncertainties (+6.0 and -5.0°F) applied, to determine the most limiting case. Therefore, analyses were performed for initial average temperatures of 595.1°F (589.1 + 6.0°F), 584.1°F (589.1 - 5.0°F), 577.0°F (571.0 + 6.0°F), and 566.0°F (571.0 - 5.0°F). For both the Loss of Normal Feedwater and the Loss of Offsite Power events, the most limiting case assumed an initial average temperature of 584.1°F.

RAI #64

Assumption 5 of Section 6.3.2.4.2 indicates that the MSIVs are assumed to be closed six seconds after receipt of a safety injection signal due to low steam line pressure (435 psia). Table 6.3.2.4-1 indicates that the low steam line pressure setpoint is reached at 0.54 seconds, in the faulted loop, and the safety injection signal isn't generated until two seconds later. Based on the above, shouldn't the MSIVs close at 8.54 seconds, rather than the 6.54 seconds indicated in the table?

FPL Energy Seabrook Response:

The words "safety injection signal due to a" are deleted from LAR Attachment 1, Section 6.3.2.4.2 "Method of Analysis" Assumption 5 (page 6-82). Main steam isolation valve closure was modeled to occur at 6.0 seconds after receipt of a low steam line pressure signal. Table 3.3-3 of the Seabrook Station Technical Specifications confirms that steam line isolation is actuated off of the low steam line pressure signal.

RAI #65

Explain the small oscillations in the steam pressure of the faulted SG (Figure 6.3.2.4-8), beginning at about 500 seconds. Is the SG dry, or nearly dry at this time?

FPL Energy Seabrook Response:

These small oscillations as the steam generator approaches dryout occur well past the time of the peak heat flux, and are due to an oscillating heat transfer coefficient within the tube region; the heat transfer mode is oscillating between nucleate boiling and forced convection vaporization. This is coincident with an oscillating void fraction that is trending upwards.

RAI #66

Assumption 9 of Section 6.3.2.4.2 indicates that the feedwater isolation valves are assumed to be closed 12 seconds after receipt of a safety injection signal due to low steam line pressure (435 psia). Table 6.3.2.4-1 indicates that the low steam line pressure setpoint is reached at 0.54 seconds in the faulted loop, and the safety injection signal isn't generated until two seconds later. Based on the above, shouldn't the feedwater isolation valves close at 14.54 seconds, rather than the 12.54 seconds indicated in the table?

FPL Energy Seabrook Response:

LAR Attachment 1, Section 6.3.2.4.2 "Method of Analysis" Assumption 9 (page 6-92) states that the feedwater isolation occurs 12.0 seconds after the steam line pressure in the faulted loop reaches the low setpoint signal that generates the safety injection signal. This is consistent with the definition of an engineered safeguards feature response time as provided in the Seabrook Station Technical Requirements Manual, which is defined as the time interval from when a monitored parameter exceeds its actuation setpoint at the channel sensor until the engineered safeguards feature equipment is capable of performing its intended safety function. With respect to feedwater isolation following a steam line break, the applicable monitored parameter is steam line pressure, which is processed with lead/lag time constants of 50 seconds and 5 seconds, respectively. Therefore, although feedwater isolation is designed to actuate off of a safety injection signal (see Table 3.3-3 of the Seabrook Station Technical Specifications), the response time is based off of the low steam line pressure signal that generates the safety injection signal. In conclusion, the analysis correctly models feedwater isolation at 12.0 seconds after the low steam line pressure signal is reached in the faulted loop.

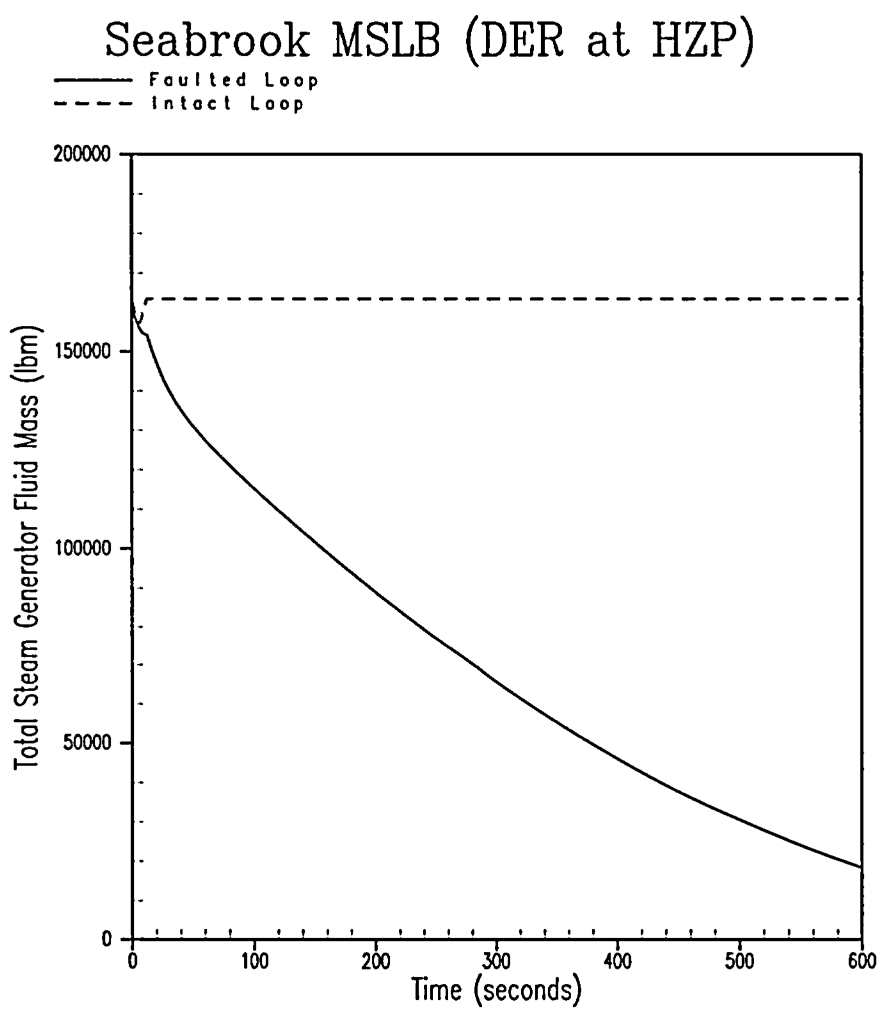
RAI #67

In support of Section 6.3.2.4, provide transient plots of inventory in the faulted and intact SGs.

FPL Energy Seabrook Response:

See Figure RAI 67-1.

FIGURE RAI-67-1
Seabrook MSLB DER - Steam Generator Mass



RAI #68

Assumption 11 of Section 6.3.3.4.2 indicates the following:

> Choked flow is assumed at the break with a high blowdown quality prior to reactor trip, resulting in an increase in the time required to obtain reactor trip. The blowdown quality after reactor trip corresponds to saturated water until the point at which all liquid inventory in the faulted steam generator is lost, resulting in a decrease in the heat removal capability of the faulted steam generator. After the liquid mass is depleted, the blowdown becomes saturated steam.

In Section 6.3.3.4.3, "Results," FPLE states the following:

> The Reactor Coolant System heatup prior to reactor trip is due to loss of the secondary system heat sink as a result of main feedwater spillage through the break and the increased secondary system temperature and pressure following the turbine trip. Reactor power increases slightly prior to the trip due to the Reactor Coolant System heatup. The primary and secondary systems were calculated to remain below 110 percent of their respective design pressures.
>
> Following the reactor trip, steam flow out the break cools the Reactor Coolant System and eventually causes Reactor Coolant System pressure to decrease and the pressurizer to empty resulting in Safety Injection initiation on a low pressurizer pressure signal. The core remains covered with water as demonstrated by the fact that the coolant loops do not reach a saturated condition. Low main steam line pressure causes closure of the main steam isolation valves and ends the cooldown period. Addition of safety injection flow aids in cooling down the primary and ensures that sufficient fluid exists to keep the core covered with water."

Prior to reactor trip, the high quality break flow, which is assumed in order to delay the time of reactor trip, would tend to cool the RCS, like a steam line break. However, the results refer to a heatup prior to reactor trip. Following reactor trip, the results discuss the cooling effect of steam flow; but there would be no steam flow out the break until the SG dries out. Please resolve these discrepancies.

FPL Energy Seabrook Response:

The inclusion of LAR Attachment 1, Subsection 6.3.3.4.2 "Method of Analysis" Assumption 11 (page 6-114) was in error. LAR Attachment 1, Subsection 6.3.3.4.2 is revised to delete assumption 11, since is not applicable to the analyses presented in the LAR. For the analyses performed in support of the Seabrook Station SPU, the blowdown quality throughout the transient is calculated by the RETRAN code. As demonstrated in the FPL Energy Seabrook response to RAI #73, the break flow prior to reactor trip consists of only water (0% quality), with increasing steam quality following reactor trip. The RETRAN model has been shown to compare well with more detailed steam generator model predictions using the NOTRUMP code (WCAP-9236 and WCAP-10079-P-A) early in the transient, but overpredicts the entrainment of break flow once the steam generator feedring uncovers. The comparisons between the LOFTRAN and RETRAN models for the feedline break event documented in Section 5.2.5.3 of WCAP-14882-P-A demonstrated that the RETRAN calculated steam generator mass/inventory is conservative relative to the LOFTRAN steam generator mass/inventory for a given trip setpoint. The Westinghouse RETRAN model has been approved for feedline break modeling through the safety evaluation report on WCAP-14882-P-A.

RAI #69

Section 6.3.3.4, "Feedwater System Line Break," indicates that the reactor trip signal is obtained from low SG level in the broken SG. How is the water level indication modeled?

FPL Energy Seabrook Response:

A description of the method used by RETRAN to calculate steam generator level is provided in Section 3.8.2 of WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses."

RAI #70

In support of Section 6.3.3.4, discuss the effects of assuming more entrainment in the break flow, prior to reactor trip, such that the reactor trip might be generated earlier when the RCS is hotter.

FPL Energy Seabrook Response:

As noted in FPL Energy Seabrook response to RAI #68, the break flow prior to reactor trip is saturated water, therefore, entrainment is 100%.

RAI #71

In support of Section 6.3.3.4, provide the assumptions and models pertaining to feedwater line break flow quality.

FPL Energy Seabrook Response:

As noted in FPL Energy Seabrook response to RAI #68, the blowdown quality throughout the transient is calculated by the RETRAN code. The RETRAN model has been shown to compare well with more detailed steam generator model predictions using the NOTRUMP code (WCAP-9236 and WCAP-10079-P-A) early in the transient, but overpredicts the entrainment of break flow once the steam generator feedring uncovers. The comparisons between the LOFTRAN and RETRAN models for the feedline break event documented in Section 5.2.5.3 of WCAP-14882-P-A demonstrated that the RETRAN calculated steam generator mass/inventory is conservative relative to the LOFTRAN steam generator mass/inventory for a given trip setpoint. It is noted that the Westinghouse RETRAN model has been approved for feedline break modeling through the safety evaluation report on WCAP-14882-P-A

RAI #72

In support of Section 6.3.3.4, discuss the expected results of analyses of a spectrum of feedwater line break sizes, over a range of assumed levels of water entrainment in the break flows.

FPL Energy Seabrook Response:

Sensitivity studies encompassing a spectrum of feedwater line break sizes and levels of water entrainment are documented in Sections 5.C.15 and 5.C.16 of WCAP-9230 "Report on the Consequences of a Postulated Main Feedline Rupture, Proprietary."

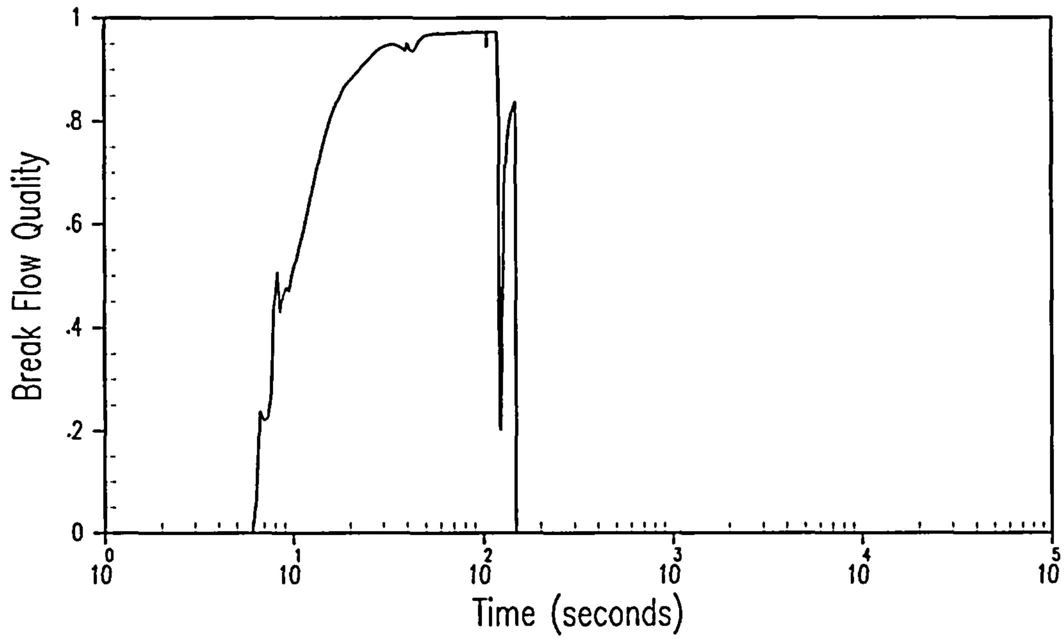
<u>**RAI #73**</u>

In support of Section 6.3.3.4, provide a transient plot of break flow quality for the feedwater line break.

<u>**FPL Energy Seabrook Response:**</u>

See Figure RAI 73-1 . The break flow quality decreases to 0.0 at ~200 seconds because the break flow ceases at this time as a result of the steam generator depressurizing to atmospheric pressure.

FIGURE RAI-73-1
Feedwater Line Break
Break Flow Quality



RAI #74

In Section 6.3.5.3.2, "Method of Analysis," FPLE states the following:

> A generic statepoint analysis for this event [Reference 6.3-25], which was performed in 1986 to bound a number of four-loop pressurized water reactors, was evaluated and determined to remain applicable for the SPU. With the generic statepoints being applicable, the effects of the SPU are accounted for when performing the nuclear and departure from nucleate boiling analyses, which are performed on a cycle-specific basis.

Explain how it was determined that the statepoints were to remain applicable for Seabrook.

FPL Energy Seabrook Response:

As described in WCAP-11394-P-A, LAR Reference 6.3-25 (page 6-374), the statepoints are a function of the rod control system characteristics, core reactivity coefficients, dropped rod cluster control assembly worths, and rod cluster control assembly control bank worth. The Seabrook Station rod control system and control rod characteristics for the SPU continue to be represented by the generic analysis. Also, the bounding ranges of dropped rod worths, control bank worths, and reactivity feedback conditions (moderator temperature coefficients) assumed in the generic statepoint analysis remain applicable to the Seabrook Station SPU. Therefore, the generic statepoints remain applicable for the SPU. Note that the statepoints represent the relative transient responses to various dropped rod scenarios, and are not sensitive to variations in initial conditions such as power, T_{avg}, and pressure.

RAI #75

Table 6.3.3.4-1 indicates that the MSIVs close only four seconds after the low steamline pressure isolation setpoint is reached; whereas, Assumption 5 of Section 6.3.2.4.2 indicates that the MSIVs are assumed to be closed six seconds after receipt of a safety injection signal. Please explain the difference.

FPL Energy Seabrook Response:

LAR Attachment 1, Table 6.3.3.4-1(page 6-121) is in error. LAR Attachment 1, Table 6.3.3.4-1 is revised to replace the words "Low Steamline Isolation Setpoint Is Reached" at 114.5 seconds with "Steamline Isolation Signal" at 112.5 seconds. Also, "All Main Steamline Isolation Valves are Closed" is clarified by adding the words "Due to Low Steam Pressure". Therefore, there is a delay of six seconds between the steamline isolation signal and the main steam isolation valve closure.

RAI #76

Tables 6.3.4.1-1 and 6.3.4.1-2 indicate that two trip time delays are assumed to be 1.5 seconds for the RCP undervoltage reactor trip, and 1.0 second for the under frequency reactor trip. Table 6.3.5.1-2 indicates that a 0.5 second trip delay time is assumed for the high neutron flux reactor trip. A 2.0 second delay time is assumed for other events. Provide a list of the trip time delays that are assumed for the events reported in the LAR, and their bases.

FPL Energy Seabrook Response:

The trip time delays assumed for each event are listed in LAR Attachment 1, Table 6.3.1-3 (page 6-60). However, the reactor coolant pump undervoltage and underfrequency trips for the Complete Loss of Reactor Coolant Flow were inadvertently omitted in the LAR. LAR Attachment 1, Table 6.3.1-3 is revised to include the time delays for the reactor coolant pump undervoltage and underfrequency trips for the Complete Loss of Reactor Coolant Flow. These delays are 1.5 seconds for the reactor coolant pump undervoltage reactor trip and 0.6 second for the reactor coolant pump underfrequency reactor trip. The bases for all the time delays are the response time test limits specified in Technical Requirements 1 and 2 of the Seabrook Station Technical Requirements Manual.

Additionally, the time to reach the setpoint and the trip time delay for the underfrequency trip were inadvertently reversed in LAR Attachment 1, Subsection 6.3.4.1.2 "Complete Loss of Forced Reactor Coolant Flow" and Table 6.3.4.1-2 (pages 6-157 and 6-164, respectively). LAR Attachment 1, Subsection 6.3.4.1.2 is revised to state that the time to reach the underfrequency trip setpoint at a frequency decay rate of 5 Hz/second is 1.0 second, and the trip time delay is 0.6 second. This has no impact on the results of the transient, since the analysis does not model the delays separately, but rather assumes a total rod motion delay of 1.6 seconds.

<u>RAI #77</u>

For the feedwater line break (Section 6.3.3.4.2), describe the RETRAN model that decreases SG heat transfer area as the shell side liquid inventory decreases. How does this model behave in the rapidly changing shell side environment caused by the feedwater line break?

<u>FPL Energy Seabrook Response:</u>

The steam generator heat transfer area does not actually decrease. Rather, the model for each conductor simply shifts the applicable heat transfer correlation for the changing conditions. This is generally equivalent to the LOFTRAN method of decreasing the heat transfer area. Assumption 20 of LAR Attachment 1, Subsection 6.3.3.4.2 "Method of Analysis" (page 6-115) should be revised to state that the RETRAN code automatically adjusts the heat transfer correlation for the steam generator tubes as the shell side inventory decreases.

RAI #78

In Section 6.3.3.4.3, FPLE states the following:

> Following the reactor trip, steam flow out the break cools the Reactor Coolant System and eventually causes Reactor Coolant System pressure to decrease and the pressurizer to empty resulting in Safety Injection initiation on a low pressurizer pressure signal.

Justify the modeling of steam flow (and not water) through the break, which would tend to mitigate the consequences of the feedwater line break.

FPL Energy Seabrook Response:

Following the feedwater line break and resultant turbine trip, the steam pressure in the intact steam generators is greater than that in the faulted steam generator. This causes steam from the intact steam generators to flow to the faulted steam generator and out the break until steamline isolation occurs.

RAI #79

In Section 6.3.5.3, it is stated that the single rod control cluster assembly (RCCA) withdrawal event is bounded by the rod ejection accident. Explain how a Condition IV event might bound a Condition III event.

FPL Energy Seabrook Response:

The statement in question was included in this section due to an editorial error and does not apply. LAR Attachment 1, Subsection 6.3.5.3.3 (page 6-190) is revised to delete the statement "Rod ejection accident, LAR Subsection 6.3.3.7, bounds this event." The conclusion that the number of rods with a departure from nucleate boiling ratio less than the limit does not exceed 5%, however, remains valid.

RAI #80

Section 6.3.5.7.3 provides results for the RCCA ejection analyses and indicates that the peak hot-spot fuel centerline temperatures reaches the fuel melting temperatures for several seconds during the full power cases. Describe how the extent of fuel melting is determined.

FPL Energy Seabrook Response:

Fuel melting is represented in the FACTRAN code by assuming that melting occurs over a 5°F temperature range instead of at a constant temperature. This is performed in the code by setting the value of specific heat (c_p) in this range such that c_p x 5°F is equal to the latent heat of fusion of the material. The percentage of the fuel pellet reaching melting is then calculated by FACTRAN based on the temperature above melting in each of the pellet volumetric zones represented, and the volume of the zone. For Seabrook Station, 10 fuel pellet zones were used in the FACTRAN calculation for the rod cluster control assembly ejection event. The fuel melt temperature for the initiation of the fuel melting is conservatively set to a low value by the user input to the code. It should be noted that even with the reported value of fuel pellet melting, the maximum reported radially-averaged peak fuel enthalpy at the hot spot of 163.8 cal/g was well within the Westinghouse analysis limit for this event of 200 cal/g.

<u>RAI #81</u>

In Section 6.3.6.1.4, "Conclusions," FPLE states the following:

> Analytical results show there will be no water flow through the pressurizer safety relief valves as a consequence of inadvertent operation of Emergency Core Cooling Systems during power operation provided that a minimum of 10 minutes is available for operator action to terminate Emergency Core Cooling Systems. No credit for operation of the pressurizer power operated relief valves was assumed.

a. Provide a justification that the operator will recognize the situation and act to terminate the safety injection flow in ten minutes or less.

b. Explain how the pressurizer power-operated relief valve (PORV) opening setpoint can be set to just 25 psi below the reactor trip setpoint. Normally, the PORV is set to open at 50 psi below the reactor trip setpoint, which is consistent with an uncertainty of +/-50 psi, as shown in Table 6.3.1-2.

c. The PORVs, if they open, would tend to limit the backpressure seen by the ECCS, and allow a greater ECCS flow into the RCS, and thereby decrease the pressurizer fill time. Explain how the opening of the PORVs can be considered to be credit in this transient. What are the results of this event when analyzed assuming the PORVs are available?

d. The PORV opening setpoint, 2400 psia, is not reached in this transient; but the maximum pressurizer pressure, 2378 psia, is within the pressure uncertainty, +/-50 psi, of the instrumentation. Justify the conclusion that the PORVs will not open.

e. Are the PORVs and their associated downstream discharge piping qualified for water relief?

f. Verify that the following initial conditions are the same as those assumed for the Seabrook UFSAR analyses: initial reactor power at the maximum value, and the initial pressurizer water level at its maximum value - consistent with steady-state full-power operation including allowances for calibration and instrument errors.

g. Provide information to show that RETRAN-02 pressurizer model can properly calculate pressure when the pressurizer is water-solid.

h. Describe the initial pressurizer water level assumed in the analysis of the Inadvertent actuation of safety injection at power. Confirm that this assumption is consistent with the technical specification restrictions at Seabrook.

<u>FPL Energy Seabrook Response:</u>

a. As a safety injection signal is required to initiate the transient, it will be accompanied by an alarm, which alerts the operators of the situation. Operator actions from this point forward are governed by emergency operating procedures. Based on these procedures, the operators will determine if safety injection flow is required, and if not, terminate it. Operator response time testing at the Seabrook Station has confirmed that safety injection flow can be terminated in 10.1 minutes or less. Note that this operator action time assumption is consistent with the current licensing basis and SPU analyses.

b. In reality, the pressurizer power-operated relief valves have a nominal opening setpoint that is equal to the normal high pressurizer pressure reactor trip setpoint (2400 psia), which is consistent with the current licensing basis, i.e., it was not changed for the SPU. With respect to the analysis of this transient, the relationship between the power-operated relief valve opening setpoint and the high pressurizer pressure reactor trip setpoint is irrelevant because the reactor is assumed to be tripped at the beginning of the transient.

c. The intent of the analysis is to demonstrate that water relief through the pressurizer safety valves is precluded, and therefore the power-operated relief valves are assumed to be unavailable, e.g., block valves are closed. The concern with water relief through the pressurizer safety valves is that they may fail open and result in an unisolable Reactor Coolant System. If the power-operated relief valves were available, they would not allow the pressurizer pressure to reach the set pressure of the pressurizer safety valves. Water relief through the power-operated relief valves is considered acceptable because the power-operated relief valve block valves provide backup isolation capability.

   The SPU analysis is consistent with the current licensing basis based on a review of Seabrook Station UFSAR Section 15.5.1.4 which states:

   > "Analytical results show there will be no water flow through the pressurizer safety relief valves (PSRVs) as a consequence of inadvertent operation of ECCS during power operation provided that either of two scenarios occurs:
   >
   > - There is no water flow through the PSRVs provided that at least one PORV is available for relief; or
   > - A minimum of 10.1 minutes is available for operator action to terminate ECCS and thereby prevent PSRV challenges without credit for operation of the pressurizer PORVs."

d. See the response to part c above.

e. The power-operated relief valves and their associated downstream discharge piping are qualified for water relief based on the following:

   1. As stated in Seabrook Station UFSAR Section 1.9, and consistent with the requirements of Task II.D.1 on NUREG 0737, "Clarification of TMI Action Plan Requirements", the power operated relief valves are qualified for operating conditions for design basis transients and accidents, which would include water relief.

      SBN-969 dated March 17, 1986 (Devincentis to Noonan) provided Seabrook Station's response to NUREG-0737, Task II.D.1 "Performance testing of Boiling Water Reactor and Pressurized Water Reactor relief and Safety Valves". The letter identified EPRI testing program results on valves similar to Seabrook's PORV under steam, water, preload, transition, and water seal conditions. The valve fully opened and fully closed during all of the tests. The EPRI Test Report was transmitted to the NRC by a letter, dated April 1, 1982, from D. P. Hoffamn, Chairman of the PWR and Relief Valve Test Program Subcommittee.

      Subsequent RAI responses provided by NYN-87136, dated November 23, 1987, detailed thermal-hydraulic analysis addressing water discharge conditions through the PORVs. Cases analyzed included a high pressure injection event.

Specifically, the PORVs were modeled to initially discharge steam and experience a transition to saturated water release plus a subsequent actuation during which 567°F water is discharged. The safety valves remain closed during this event. The resulting piping loads were found to be acceptable.

2. As stated in Seabrook Station UFSAR Section 3.9(B).3.3, on pages 35 and 36:

> "When the valves open, the dynamic effects from the flow of water and steam are included in the design analysis.
>
> These transient load effects on the piping system, upstream and downstream of the safety and relief valves, have been evaluated in the following manner:
>
> Pressurizer Relief Valve Piping System
>
> Both static and time history analyses were performed for the Pressurizer Relief Valve Piping System using transient loads obtained from a RELAP 5 analysis. The Pressurizer Relief Valve Piping System contains water seals and is subjected to water slugs. The effects of these two items were fully accounted for in the RELAP 5 analysis."

f. With respect to the initial reactor power, a value of 3658 MWt was assumed in conjunction with a maximum reactor coolant pump heat of 20 MWt. The total thermal power is consistent with the maximum (conservative) NSSS power of 3678 MWt (see FPL Energy Seabrook response to RAI #33 for a discussion of the power uncertainty). As for the initial pressurizer water level, a maximum value of 65% span was assumed, which corresponds to the nominal pressurizer level associated with the maximum full power T_{avg} of 589.1°F (60% span) plus an allowance of 5% span for calibration and instrument errors.

g. The qualification of the Westinghouse RETRAN-02 pressurizer model is discussed in WCAP-14882-P-A. Appendices A and B of WCAP-14882-P-A contain NRC RAIs and Westinghouse responses, respectively, in support of the qualification. See the RAI response to the staff's generic SER and TER limitations on RETRAN, in particular General Limitation item o regarding the non-equilibrium pressurizer model.

h. See the response to f. above for the assumed initial pressurizer water level (65% span). The Seabrook Station Technical Specification 3.4.3 restriction requires the pressurizer to be operable with a water volume of less than or equal to 92% of pressurizer level. The concern raised by this RAI appears to be that the plant may be operated within all existing Technical Specification Limiting Conditions for Operation and yet may be in a condition that is not consistent with the analysis performed in the Seabrook Station UFSAR. Consistent with the current licensing basis for the Seabrook Station, the initial conditions, e.g., initial pressurizer level, assumed in the accident analyses for Chapter 15 of the Seabrook Station UFSAR are based on the nominal programmed values. To these nominal values are added appropriate measurement uncertainties that are added in the conservative direction (positive or negative) for each accident.

RAI #82

For the Anticipated Transients Without Scram (ATWS) analyses, described in Section 6.3.8:

a. What value for the moderator temperature coefficient was assumed?

b. Compare the results of the Seabrook loss of load and loss of feedwater ATWS analyses, with 1760 gpm auxiliary feedwater flow, to the corresponding results of Reference 6.3-20, as adjusted for the higher power level of the Seabrook SPU.

FPL Energy Seabrook Response:

a. The assumed moderator temperature coefficient was -8 pcm/°F, which is bounding for 95% of a representative cycle.

b. Comparisons of the peak calculated Reactor Coolant System pressures are presented in Table RAI 82-1.

Table RAI 82-1:
Anticipated Transients Without Scram Results Comparison

	Peak RCS Pressure (psia)	
	Reference 6.3-20	Seabrook SPU
Loss of Normal Feedwater Anticipated Transients Without Scram	2830.4	3016.4
Loss of Load Anticipated Transients Without Scram	2901.5	3173.0

RAI #83

In Section 6.3.3.2.3, FPLE states the following:

> Following the reactor and turbine trip from full load, the water level in each steam generator falls due to the reduction of the steam generator void fraction, and because steam flow through the main steam safety valves continues to dissipate the stored and generated heat.

Level and void fraction were not explicitly calculated using LOFTRAN's one-node steam generator secondary side model. Describe how this is done in RETRAN's SG secondary side model.

FPL Energy Seabrook Response:

A description of the method used by RETRAN to calculate steam generator level is provided in WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

RAI #84

In Section 6.36.2-1, what operator actions, if any, are assumed to be implemented, and when, for the chemical and volume control system malfunction event?

FPL Energy Seabrook Response:

The Chemical and Volume Control System malfunction event analysis was performed with the assumption of no operator action. The resulting sequence of events presented in LAR Attachment 1, Table 6.3.6.2-1 (page 6-240) shows that without operator action, the pressurizer would be water-solid at about 13 minutes into the transient. Since the only operator action required is to diagnose the problem and secure the charging pumps, the inadvertent operation of the Emergency Core Cooling System during power operation, which requires more operator actions in a shorter period of time, is considered bounding.

RAI #85

In Figure 6.3.6.1-2, ECCS flow is terminated manually at ten minutes, which causes pressurizer pressure to fall by almost 200 psi. However, the pressurizer remains full for more than six minutes following operator intervention. Explain why the pressurizer water level does not fall.

FPL Energy Seabrook Response:

The pressurizer water level does not fall because the pressurizer sprays continue to inject water from the cold leg due to the pressure increase.

RAI #86

In Section 6.3.6.2-2, what causes the pressurizer pressure and vessel average temperature to peak soon after the pressurizer fills? Explain why the pressure and temperature then begin to increase, at a lower rate, through the end of the reported transient.

FPL Energy Seabrook Response:

First, note that the Chemical and Volume Control System malfunction transient is analyzed for the purpose of determining whether there is enough time for operators to terminate the transient. As such, the transient results up to the time of filling the pressurizer are of most interest. Beyond this time is not a concern because the operators are assumed to act in time to preclude pressurizer filling. However, the pressurizer pressure turns around soon after the pressurizer fills because of an increased pressurizer spray flow in response to the pressure increase. The relatively cool spray flow from the cold leg condenses steam bubbles within the pressurizer and reduces the pressure. The pressure increases again as a result of the continued injection of charging flow; the lower pressurization rate is due to the increased spray flow. With respect to the vessel average temperature, it changes by less than 1°F, which is considered negligible.

RAI #87

In Section 6.3.7.1-1, verify that the Overtemperature ΔT trip provides adequate protection during this event in cases (i.e., for gradual depressurization) where the low pressurizer pressure setpoint may not be reached for a long time. If applicable, consider a single failure consisting of operating with the wrong reference temperature in the Overtemperature ΔT trip setpoint calculation.

FPL Energy Seabrook Response:

The Overtemperature ΔT reactor trip function provides protection during the Reactor Coolant System depressurization event independent of how gradual the depressurization rate occurs because the Overtemperature ΔT setpoints are calculated assuming steady-state conditions which bound the safety analysis departure from nucleate boiling limit. The Overtemperature ΔT setpoints are encompassed by the safety analysis high and low pressurizer pressure reactor trip functions. Therefore, in the limit, that is assuming the slowest depressurization rate possible, the Overtemperature ΔT reactor trip function would ensure that the departure from nucleate boiling design basis is satisfied. If the single failure was that the wrong reference temperature was used in the Overtemperature ΔT setpoint, the analyses would not be affected since the Overtemperature ΔT protection logic is 2 out of 4 loops. During the Reactor Coolant System depressurization event, each setpoint would effectively see the same pressure as the pressure is being measured in one location (that is, in the pressurizer). Therefore, at least 3 of the 4 Overtemperature ΔT setpoints would be generated such that the departure from nucleate boiling design basis is satisfied.

RAI #88

Please note that Reference 15.1.6.5 of the Seabrook UFSAR indicates that WCAP-11397-P-A was issued in April, 1984. This should be corrected to show an issue date of April, 1989, as indicated in Reference 7.1-1 of the LAR.

FPL Energy Seabrook Response:

A UFSAR change request will be initiated to revise Seabrook Station UFSAR Reference 15.1.6.5 to correct the date to April, 1989.

RAI #89

In of the TS 4.2.2.2.g change, justify the increase in the range of applicability of the limits specified in Specifications 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f.

FPL Energy Seabrook Response:

The Westinghouse F_Q surveillance methodology was developed and approved in the early 1980s (Reference 1). At that time, the top and bottom 15% of the core was excluded from surveillance due to the low importance of these regions (relatively low power), as well as to be consistent with the FXY Surveillance Technical Specification. Since that time, changes in fuel products and fuel management (e.g., the use part length burnable absorbers, as well as enriched axial blankets) have tended to move the predicted peak F_Q closer to the 15% exclusion zone.

The Seabrook Station SPU analysis identified that some hypothetical future cycle fuel management techniques could potentially move the predicted peak F_Q into the bottom 15% of the core. An extension of the F_Q surveillance zone was determined to be prudent, in order to maintain the ability of the plant F_Q surveillance measurements to identify the peak measured F_Q. Therefore, the F_Q surveillance exclusion zone was conservatively reduced from +/-15% to +/-10%.

Reference 1 - WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification," June 1983.

RAI #90

In support of TS Table 2.2-1 changes, explain why total allowance is not applicable to steam generator low-low level setpoint.

FPL Energy Seabrook Response:

See FPL Energy Seabrook response to RAI #4.

RAI #91

In support of TS Table 3.3-4 changes, explain why total allowance is not applicable to steam generator low-low and high-high level setpoints.

FPL Energy Seabrook Response:

See FPL Energy Seabrook response to RAI #4.

<u>RAI #92</u>

With regard to TS 6.8.6.1.b changes, explain why the older report, YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station," October, 1992, is substituted for WCAP-14551-P, (Proprietary), "Westinghouse Setpoint Methodology for Protection Systems, Seabrook Nuclear Power Station Unit 1, 24 Month Fuel Cycle Evaluation," June, 1998.

<u>FPL Energy Seabrook Response:</u>

WCAP-14551-P was specifically issued for the 24 month fuel cycle program which was never implemented by Seabrook Station, and therefore, this supporting WCAP was not submitted for NRC review. For this reason, it was removed from the listed references in Technical Specification 6.8.6.1.b. Report YAEC-1854P is not considered a substitute for WCAP-14551-P, but was retained in Technical Specification 6.8.6.1.b as an applicable reference.

Human Performance

RAI #93

Table Matrix-11 of NRC Review Standard RS-001, Revision 0, provides the NRC staff's guidance for evaluating the potential impacts for SPUs on human performance issues and outlines specific review questions. Section 11.0, "Impact on Operations," of the LAR, does not sufficiently address the standard set of questions of Matrix-11.

a. Describe how the proposed SPU will change the plant emergency and abnormal operating procedures.

b. Provide a list of systems that require new operating and maintenance procedures as a result of the SPU. Provide a description of each of the new procedures's intended purpose.

c. Describe any new operator actions required as a result of the SPU. Describe changes to current operator actions related to emergency or abnormal operating procedures that will occur as a result of the SPU.

d. Describe any changes the SPU will have on the safety parameter display system. How will the operators be aware of such changes?

e. Describe any changes the SPU will have on the operator training program and the plant referenced control room simulator. State the controlling standard for the control room simulator. Provide an implementation schedule for modifications to the simulator that are needed as a result of the SPU.

FPL Energy Seabrook Response:

a. The changes are as follows:

Emergency Operating Procedures

- The maximum allowed hot leg recirculation switchover time is being reduced from 9 hours to 7 hours and a minimum time of 5.5 hours has been established.
- Emergency operating procedure setpoint values are being changed to support the SPU. For example; Reactor Coolant System hot leg temperature equivalent to saturation pressure, pressurizer levels, and steam generator levels.
- The minimum Emergency Core Cooling System flow versus decay heat removal curve will be modified.
- Natural circulation cooldown emergency operating procedures are being changed to delay entry into functional restoration guideline (FR-H.1), which deals with the loss of secondary heat sink. This change is being done to support the extended time for natural circulation cooldown to achieve Residual Heat Removal System cut-in conditions as described in LAR Attachment 1, Section 10.2 (page 10-2).

Abnormal Operating Procedures

- Applicable NSSS instrument failure abnormal operating procedures are being changed to support new SPU setpoints.
- Applicable balance of plant instrument failure abnormal operating procedures are being changed to support new SPU setpoints.
- Applicable balance of plant component malfunction abnormal operating procedures are being changed to support condensate pump impeller upgrades.
- Applicable abnormal operating procedures are being changed to reflect the new megawatt thermal value.
- Applicable abnormal operating procedures are being revised with new SPU calorimetric values.
- Applicable abnormal operating procedures are being enhanced to allow for higher capacity demineralized water storage tank gravity feed makeup to the condensate storage tank.
- Shutdown Loss of Coolant Accident abnormal operating procedure is being changed to support the new SPU setpoints.
- Applicable NSSS component malfunction abnormal operating procedures are being changed to support new SPU pressurizer level band setpoints.
- Applicable balance of plant component malfunction abnormal operating procedures are being changed to support the new SPU steam generator manual level control band.
- Loss of Residual Heat Removal abnormal operating procedures are being changed to update decay heat removal and time to boil curves.

b. Basic system operation and monitoring will not be affected by the SPU. There are no new systems required by the SPU. There are no new operating and maintenance procedures required by the SPU. There will be several one-time use procedures created for the SPU post-outage power ascension. Specifically, one-time use procedures will include such subjects as tenth refueling outage uprate power ascension testing (administrative controls), turbine–generator performance testing and moisture–separator reheater testing. There are numerous plant procedures that will require revision to reflect the SPU. The total estimated number is in excess of 200.

Aside from the one-time power ascension procedures, there are no new operating and maintenance procedures required by the SPU. The one-time use procedures are intended to control the power ascension and test the uprated plant in a safe and conservative manner. The balance of plant performance testing will be carried out to confirm the actual thermal and electrical plant secondary side parameters are consistent with engineering predictions.

c. No new operator actions are required by the SPU. Changes to current operator actions are as follows:

Emergency Operating Procedures

- The maximum hot leg recirculation switchover time is being reduced from 9 hours to 7 hours and a minimum time of 5.5 hours has been established. (See FPL Energy Seabrook Response to RAI #41.)

Abnormal Operating Procedures

- No changes to operator actions in abnormal operating procedures are required.

d. No changes to the layout, monitoring, or use of the Safety Parameter Display System are required to support the SPU. Revisions to several of the setpoints used in the Safety Parameter Display System status trees are anticipated. For example, the setpoint used by Safety Parameter Display System to indicate that steam generator narrow range level is "on span" may increase slightly above its current value. The operator actions indicated by the Safety Parameter Display System in response to a narrow range level below this setpoint are not changing; however, the setpoint directing the operator to initiate the action may increase.

The Operations Department has been integrated into the uprate process. An Operations Department representative joined the uprate team at an early stage. The design change process requires Operations Department reviews and signoffs on the design change packages. In addition, presentations of the SPU design and scope have been made to all licensed operators as part of licensed operator requalification training classes.

e. The simulator will be upgraded in both hardware and software to match the SPU design. Examples include re-banding of the main steam and feedwater flow indicators to reflect the increase in steam and feedwater flow and modification to the generator megawatt meter to accommodate the increased megawatt output. The simulator core model and secondary plant models will be revised based upon SPU design data. These changes will be incorporated into the simulator prior to implementation in the plant to allow for operator familiarization training. Licensed and non-licensed operator training on power uprate modifications will be conducted in the second phase of operator continuing training in 2005.

The controlling standard for the simulator is ANSI/ANS 3.5-1998.

The modifications to the simulator will be completed by the end of January 2005 to allow for operator training on the power uprate modifications in the second phase of continuing training which begins in mid-February 2005. Licensed operators will be trained on the modified simulator prior to the 2005 outage.

Environmental Impact

RAI #94

In Section 13.0, "Environmental Evaluation," FPLE states the following:

> The recirculation mode increases this discharge water temperature and therefore, the temperature rise between the intake and discharge transition structure is also increased.

a. Does the National Pollutant Discharge Elimination System (NPDES) permit for Seabrook require a mixing zone? Is there adequate mixing of the thermal plume to accommodate the increase in circulating water outlet temperature.

b. Will the SPU require any changes to the current NPDES permit or other plant administrative limits?

c. Discuss the noise effects due to operation of Seabrook at uprated power conditions. Will there be an increase in noise resulting from the SPU?

FPL Energy Seabrook Response:

a. Seabrook Station's National Pollutant Discharge Elimination System Permit defines a "near-field jet mixing region" as "that portion of the receiving waters within 300 feet of the submerged diffuser in the direction of discharge" and states "The thermal component of the discharge from the Seabrook Station shall not cause a monthly mean temperature rise of more than 5°F in the near-field jet mixing region." Compliance with this temperature limit is reported annually to Environmental Protection Agency. Historical data indicates that maximum monthly mean temperatures of 4°F have been reported. As stated in LAR Attachment 1, Table 13.2-1 (page 13-3), the projected maximum monthly mean temperature projection under SPU conditions is 4.24°F. Therefore, the 5°F permit requirement will continue to be met. No change to this National Pollutant Discharge Elimination System Permit limit is required for the SPU .

b. No changes to the current National Pollutant Discharge Elimination System Permit thermal limits or any other parameter, or administrative requirements are anticipated to support the SPU.

c. LAR Attachment 1, Section 13.1 (page 13-1) stated:

> "The Final Environmental Statement concluded that, after weighing the environmental, economic, technical, and other benefits, against environmental costs and considering available alternatives, the issuance of an operating license was an acceptable action. The Final Environmental Statement conclusions are not impacted as a result of the SPU."

The Final Environmental Statement makes reference to noise considerations and noted the acceptability of noise levels at the site boundary.

The Seabrook Station SPU did not require any new and large motors or pumps. Consideration of other features affected by the SPU did not reveal any new and significant sources of noise that would be expected to be noticeable at the site boundary. Therefore, in the matter of noise, the conclusions of the Final Environmental Statement remain unchanged.

RAI #95

Verify that for post-accident conditions, the existing post-accident dose rate maps are adequate for power uprate conditions.

FPL Energy Seabrook Response:

The SPU evaluation concluded that the existing post-accident dose rate maps are adequate at SPU conditions.

As stated in LAR 04-03, Reference 8.4.2 of Attachment 1, Subsection 8.4.15.2.2 (page 8-103), compliance with NUREG 0737 II.B.2 for the current licensed power level included development of post-LOCA radiation zone dose rate maps used for planning of post accident operations.

In accordance with LAR Reference 8.4.2, the post-LOCA radiation dose rate maps were developed to provide an indication of the radiation levels at plant vital areas, and areas essential to access those vital areas for purposes of post-accident mitigation "planning." Each zone in the dose rate maps represents a range of dose rates covering a decade (e.g., 10 E02 to 10 E03 mrem/hr).

As noted in LAR Reference 8.4.2, in developing these maps, conservative assumptions were made in determining the dose rates from the various post-accident sources. For example, dose rate estimates were based on simplified, but conservative models of actual plant configurations. Dose rates were calculated at the midpoint of each source, regardless of the relative elevation of the source and receptor. The above approach conservatively ignored actual distances or increased effective shielding due to slant path through the shield.

The SPU evaluation concluded that the percentage increase in source terms between the currently analyzed basis (core power level of 3565 MWt), and the analyzed SPU core power level of 3659 MWt, is well within the conservatism of the radiation dose rate zone boundaries depicted in the maps. This conclusion is based on conservative modeling techniques utilized to establish the estimated dose rate range applicable to a zone, and the typical conservative approach utilized when establishing the approximate location of zone boundaries.

In summary, the SPU evaluation has concluded that the existing post-accident dose rate maps are adequate to support its intended function of providing "indication of radiation levels" during post-accident mitigation "planning," at SPU conditions.

Containment Assessment

RAI #96

In support of LAR Section 6.4.1, "Loss of Coolant Accident (LOCA) Mass and Energy Release," justify the use of equipment qualification temperatures for the acceptance criterion for containment LOCA calculations. Why is this acceptable to demonstrate structural adequacy?

FPL Energy Seabrook Response:

The text in LAR Attachment 1, Subsection 6.4.1 "Loss of Coolant Accident (LOCA) Mass and Energy Release" (page 6-255) is not intended to imply that the containment structural integrity is demonstrated through adherence to the equipment qualification envelope. Containment structural integrity is demonstrated by the fact that the peak containment pressure following LOCA under SPU conditions is less than the established containment design pressure and less than the existing analysis of record. Additionally, the peak temperature obtained for SPU conditions post-LOCA is less than the peak temperature obtained for the existing analysis of record and is bounded by the existing equipment qualification temperature profile.

RAI #97

In Section 6.4.1.1.7, "Acceptance Criteria for Analyses," FPLE states that the criteria for sources of heat for the LOCA mass and energy release calculations is stated as Appendix K Paragraph I.A. However, the decay heat is stated in 6.4.1.1.8, "Mass and Energy Release Data," as being calculated from Reference 6.4-5, "American National Standard for Decay Heat Power in Light Water Reactors." These are different models. Please explain the inconsistency.

FPL Energy Seabrook Response:

10 CFR 50 Appendix K, Paragraph I.A addresses several sources of energy, including: 1) Initial reactor power and calorimetric uncertainty, 2) Initial stored energy in the fuel, 3) Fission heat, 4) Decay of the Actinides, 5) Fission product decay, 6) Metal water reaction rate, 7) Reactor internals heat transfer, and 8) Pressurized water reactor primary to secondary heat transfer. The intent of the statement in LAR Attachment 1, Subsection 6.4.1.1.7 (page 6-261) is to indicate that these sources (i.e., those listed in 10 CFR 50, Appendix K, Paragraph I.A) have been considered in the analyses. The actual models used are described in the LAR. As stated in LAR Attachment 1, Subsection 6.4.1.1.8 (page 6-264), the NRC has approved the use of the American National Standard-5.1, November 1979 decay heat model for the calculation of energy releases to the containment following a LOCA.

RAI #98

In Section 6.4.4.2, "Input Parameters and Assumptions," and Section 6.4.4.4, "Results," FPLE gives Reference 6.4-1, "Westinghouse LOCA Mass and Energy Release Model for Containment Design -March 1979 Version," as a reference for main steam line break mass and energy release guidance. Reference 6.4-1 refers to LOCAs. Verify that this is the correct reference.

FPL Energy Seabrook Response:

LAR Attachment 1, Reference 6.4-1 (page6-374) should be the following:

> WCAP-8822 (Proprietary) and WCAP-8860 (Non-proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Non-proprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Non-proprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Sub-atmospheric Containment Designs," September 1986.

RAI #99

State the version of the GOTHIC computer code was used for the Seabrook containment analysis. Verify that GOTHIC was used consistent with the NRC safety evaluation report for Kewaunee dated September 29, 2003.

FPL Energy Seabrook Response:

The GOTHIC code version 7.0p2 (QA) was used for the Seabrook Station SPU analysis. NRC Safety Evaluation Report for the Kewaunee Nuclear Station (Amendment 160 to Docket No. 50-305, letter to Mr. Thomas Couto, dated September 23, 2003) was consulted in detail and the Seabrook Station analysis is consistent with that Safety Evaluation Report. The Seabrook Station containment evaluation does not utilize the Gido-Koestel heat transfer model, nor does it use the Mist-Diffusion Layer Model (MDLM) options of GOTHIC. Passive heat transfer to containment structures was modeled using standard and accepted Tagami and Uchida heat transfer correlations. No scaling height factor (λ_h) was employed in the analysis. Recommended and appropriate break effluent and spray droplet sizes were employed and nitrogen injection to the containment from the accumulators was conservatively modeled.

In addition to the Kewaunee Safety Evaluation Report, the Fort Calhoun Station Safety Evaluation Report (Amendment 222 to Docket No. 50-285, letter to Mr. R.T Ridenoure, dated November 5, 2003) and other GOTHIC-related Safety Evaluation Reports and Requests for Additional Information were addressed in the development of the analysis.

Attachment RAI 2-1

Table 1
Steam Generator Water Level - High-High

Parameter | **Allowance*** [ ]$^{+a,c}$

Process Measurement Accuracy
[ ]$^{+a,c}$ (PMA$_{PP}$)
[ ]$^{+a,c}$ (PMA$_{RL}$)
[ ]$^{+a,c}$ (PMA$_{PD}$)
[ ]$^{+a,c}$ (PMA$_{FV}$)
[ ]$^{+a,c}$ (PMA$_{DL}$)
[ ]$^{+a,c}$ (PMA$_{SC}$)
[ ]$^{+a,c}$ (PMA$_{ID}$)
[ ]$^{+a,c}$ (PMA$_{FR}$)
[ ]$^{+a,c}$ (PMA$_{MD}$)

Primary Element Accuracy (PEA)

Sensor Calibration Accuracy (SCA)

Sensor Reference Accuracy (SRA)

Measurement & Test Equipment Accuracy (SMTE)

Sensor Pressure Effects (SPE)

Sensor Temperature Effects (STE)

Sensor Drift (30 months) (SD)

Bias
[ ]$^{+a,c}$ (Bias$_1$)
Systematic Pressure Effect (Bias$_5$)

Rack Calibration
Rack Accuracy (RCA)
Reference Accuracy (RRA)
Measurement & Test Equipment Accuracy (RMTE)

Rack Temperature Effect (RTE)

Rack Drift (RD)

---

* In percent span (0-100% Narrow Range Level, 85.72 inches)

Table 1 (Continued)
Steam Generator Water Level - High-High

Channel Statistical Allowance =

+a,c

[ ]

+a,c

[ ]

Table 2
Steam Generator Water Level - Low-Low
Loss of Normal Feedwater Analysis

Parameter	Allowance*
Process Measurement Accuracy	+a,c
[ ]+a,c (PMA_{PP})	
[ ]+a,c (PMA_{RL})	
[ ]+a,c (PMA_{SC})	
[ ]+a,c (PMA_{FV})	
[ ]+a,c (PMA_{DL})	
[ ]+a,c (PMA_{ID})	
[ ]+a,c (PMA_{FR})	
[ ]+a,c (PMA_{MD})	
Primary Element Accuracy (PEA)	
Sensor Calibration Accuracy (SCA)	
Sensor Reference Accuracy (SRA)	
Measurement & Test Equipment Accuracy (SMTE)	
Sensor Pressure Effects (SPE_L)	
Sensor Temperature Effects (STE)	
Sensor Drift (30 months) (SD)	
Bias	
[ ]+a,c ($Bias_1$)	
Systematic Pressure Effect ($Bias_3$)	
Rack Calibration	
Rack Accuracy (RCA)	
Reference Accuracy (RRA)	
Measurement & Test Equipment Accuracy (RMTE)	
Rack Temperature Effect (RTE_B)	
Rack Drift (RD_B)	

* In percent span (0-100% Narrow Range Level, 85.72 inches)

Table 2 (Continued)
Steam Generator Water Level - Low-Low
Loss of Normal Feedwater Analysis

Channel Statistical Allowance =

+a,c

[

+a,c

]

[

]

Table 3
Steam Generator Water Level - Low-Low
Large Feedline Break Analysis

Parameter	**Allowance***
	$[\quad]^{+a,c}$
Process Measurement Accuracy	
[$]^{+a,c}$ (PMA$_{PP}$)	
[$]^{+a,c}$ (PMA$_{RL}$)	
[$]^{+a,c}$ (PMA$_{FV}$)	
[$]^{+a,c}$ (PMA$_{DL}$)	
[$]^{+a,c}$ (PMA$_{SC}$)	
[$]^{+a,c}$ (PMA$_{ID}$)	
[$]^{+a,c}$ (PMA$_{FR}$)	
[$]^{+a,c}$ (PMA$_{MD}$)	
Primary Element Accuracy (PEA)	
Sensor Calibration Accuracy (SCA)	
Sensor Reference Accuracy (SRA)	
Measurement & Test Equipment Accuracy (SMTE)	
Sensor Pressure Effects (SPE)	
Sensor Temperature Effects (STE)	
Sensor Drift (30 months) (SD)	
Environmental Allowance	
Transmitter Adverse Temperature Effects (EA$_1$)	
Cable IR Effects (Bias$_2$)	
Reference Leg Heatup (EA$_3$)	
Bias	
[$]^{+a,c}$ (Bias$_1$)	
Systematic Pressure Effect (Bias$_3$)	
Rack Calibration	
Rack Accuracy (RCA)	
Reference Accuracy (RRA)	
Measurement & Test Equipment Accuracy (RMTE)	
Rack Temperature Effect (RTE)	
Rack Drift (RD)	

* In percent span (0-100% Narrow Range Level, 85.72 inches)

Table 3 (Continued)
Steam Generator Water Level - Low-Low
Large Feedline Break Analysis

Channel Statistical Allowance =

+a,c

[ ]

+a,c

[ ]

Table 4
Steam Generator Water Level - Low-Low
Small/Intermediate Feedline Break Analysis

Parameter	**Allowance*** $^{+a,c}$
Process Measurement Accuracy	
[$]^{+a,c}$ (PMA_{PP})	
[$]^{+a,c}$ (PMA_{RL})	
[$]^{+a,c}$ (PMA_{FV})	
[$]^{+a,c}$ (PMA_{DL})	
[$]^{+a,c}$ (PMA_{SC})	
[$]^{+a,c}$ (PMA_{ID})	
[$]^{+a,c}$ (PMA_{FR})	
[$]^{+a,c}$ (PMA_{MD})	
Primary Element Accuracy (PEA)	
Sensor Calibration Accuracy (SCA)	
Sensor Reference Accuracy (SRA)	
Measurement & Test Equipment Accuracy (SMTE)	
Sensor Pressure Effects (SPE)	
Sensor Temperature Effects (STE)	
Sensor Drift (30 months) (SD)	
Environmental Allowance	
Transmitter Adverse Temperature Effects (EA_1)	
Cable IR Effects ($Bias_2$)	
Reference Leg Heatup (EA_2)	
Bias	
[$]^{+a,c}$ ($Bias_1$)	
Systematic Pressure Effect ($Bias_3$)	
Rack Calibration	
Rack Accuracy (RCA)	
Reference Accuracy (RRA)	
Measurement & Test Equipment Accuracy (RMTE)	
Rack Temperature Effect (RTE)	
Rack Drift (RD)	

* In percent span (0-100% Narrow Range Level, 85.72 inches)

Table 4 (Continued)
Steam Generator Water Level - Low-Low
Small/Intermediate Feedline Break Analysis

Channel Statistical Allowance =

+a,c

[ ]

+a,c

[ ]

Table 5
Steam Generator Water Level - Low-Low
Steam Break Analysis

Parameter	**Allowance*** $^{+a,c}$
Process Measurement Accuracy	
[ ]$^{+a,c}$ (PMA_{PP})	
[ ]$^{+a,c}$ (PMA_{RL})	
[ ]$^{+a,c}$ (PMA_{FV})	
[ ]$^{+a,c}$ (PMA_{DL})	
[ ]$^{+a,c}$ (PMA_{SC})	
[ ]$^{+a,c}$ (PMA_{ID})	
[ ]$^{+a,c}$ (PMA_{FR})	
[ ]$^{+a,c}$ (PMA_{MD})	
Primary Element Accuracy (PEA)	
Sensor Calibration Accuracy (SCA)	
Sensor Reference Accuracy (SRA)	
Measurement & Test Equipment Accuracy (SMTE)	
Sensor Pressure Effects (SPE)	
Sensor Temperature Effects (STE)	
Sensor Drift (30 months) (SD)	
Environmental Allowance	
Bias	
[ ]$^{+a,c}$ ($Bias_1$)	
Systematic Pressure Effect ($Bias_3$)	
Rack Calibration	
Rack Accuracy (RCA)	
Reference Accuracy (RRA)	
Measurement & Test Equipment Accuracy (RMTE)	
Rack Temperature Effect (RTE)	
Rack Drift (RD)	

* In percent span (0-100% Narrow Range Level, 85.72 inches)

Table 5 (Continued)
Steam Generator Water Level - Low-Low
Steam Break Analysis

Channel Statistical Allowance =

+a,c

[ ]

+a,c

[ ]

Table 6
Overtemperature ΔT Reactor Trip

Parameter **Allowance***

Process Measurement Accuracy

[ ] +a,c [ ] +a,c

Primary Element Accuracy (PEA)

Sensor Calibration Accuracy

[ ] +a,c

Sensor Reference Accuracy

[ ] +a,c

Sensor Measurement & Test Equipment

[ ] +a,c

Sensor Pressure Effects (SPE_P)

Sensor Temperature Effects

[ ] +a,c

Sensor Drift

[ ] +a,c

Table 6 (Continued)
Overtemperature ΔT Reactor Trip

Parameter	Allowance*
Bias	
[ ]+a,c	[ ]+a,c
Environmental Allowance	
Seismic (Rack)	
[ ]+a,c	
Rack Calibration Accuracy	
[ ]+a,c	
Rack Measurement & Test Equipment Accuracy	
[ ]+a,c	
Rack Temperature Effect	
[ ]+a,c	

Table 6 (Continued)
Overtemperature ΔT Reactor Trip

Parameter		**Allowance***
Rack Drift		
[ ] +a,c		[ ] +a,c
α - Accuracy of hot leg streaming [ ]+a,c		
R/E nonlinearity (RE_{LIN})		
RTD Lead Imbalance (RTD_{li})		

---

* In percent ΔT span (T_{avg} – 100 °F, pressure - 900 psi, power – 150 % RTP, ΔT - 84.8 °F = 150 % RTP, ΔI – 120 % ΔI)
** See Table 7 for gain and conversion calculations

Table 6 (Continued)
Overtemperature ΔT Reactor Trip

\# Hot Leg RTDs = 2/Loop (1 RTD assumed failed)
@Cold Leg RTDs = 1/Loop

Channel Statistical Allowance =

+a,c

+a,c

Table 7
Overtemperature ΔT Calculations

The equation for Overtemperature ΔT is:

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{1}{(1+\tau_3 S)} \leq \Delta T_0 \{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} [\qquad] + K_3 (P - P') - f_1(\Delta I)\}$$

K_1 (nominal)	=	1.21 Technical Specification value
K_1 (max)	=	[ ]+a,c
K_2	=	0.021 Technical Specification value
K_3	=	0.0011 Technical Specification value
vessel ΔT	=	56.5 °F smallest ΔT based on evaluation of temperature data
ΔI gain	=	1.71 % Technical Specification value

PMA conversions:

ΔI I/E mismatch (PMA_2)	=	[
ΔI Incore flux (PMA_3)	=	
ΔT Burndown (PMA_5)	=	
Power Cal. (PMA_4)	=	]+a,c

Pressure conversions:

Pressure gain	=	[
Pressure (SCA_P)	=	
Pressure (SRA_P)	=	
Pressure ($SMTE_P$)	=	
Pressure (STE_P)	=	
Pressure (SD_P)	=	
Pressure ($Bias_{P1}$)	=	
Pressure ($Bias_{P2}$)	=	
Pressure ($RMTE_P$)	=	
Pressure (NPC_P)	=	]+a,c

Table 7 (Continued)
Overtemperature ΔT Calculations

ΔI conversions:

ΔI conversion =
ΔI ($RMTE_{\Delta I}$) =
ΔI ($Seis_{\Delta I}$) =

[ ]+a,c

NIS conversions:

NIS ($RMTE_{NIS}$) =
NIS (RTE_{NIS}) =

[ ]+a,c

T_{avg} conversions:

T_{avg} conversion =
T_{avg} ($RMTE_{Tavg}$) =

[ ]+a,c

Total Allowance = [ ]+a,c = 8.7 %

Table 8
Overpower ΔT Reactor Trip

Parameter	Allowance*
Process Measurement Accuracy	+a,c
+a,c	
Primary Element Accuracy (PEA)	
Sensor Calibration Accuracy	
+a,c	
Sensor Reference Accuracy	
+a,c	
Sensor Measurement & Test Equipment	
+a,c	
Sensor Pressure Effects	
Sensor Temperature Effects	
Sensor Drift	
+a,c	
Environmental Allowance	
+a,c	
Rack Calibration Accuracy	
+a,c	

Table 8 (Continued)
Overpower ΔT Reactor Trip

Parameter	Allowance*
Rack Measurement & Test Equipment Accuracy	
[ ]+a,c	[ ]+a,c
Rack Temperature Effect	
[ ]+a,c	
Rack Drift	
[ ]+a,c	
α - Accuracy of hot leg streaming [ ]+a,c	
R/E nonlinearity (RE_{LIN})	
RTD Lead Imbalance (RTD_{li})	

---

* In percent ΔT span (Tavg - 100 °F, power - 150 % RTP, ΔT - 84.8 °F = 150 % RTP)
** See Table 9 for gain and conversion calculations

Table 8 (Continued)
Overpower ΔT Reactor Trip

Hot Leg RTDs = 2/Loop (1 RTD assumed failed)
@Cold Leg RTDs = 1/Loop

Channel Statistical Allowance =

+a,c

+a,c

Table 9
Overpower ΔT Calculations

The equation for Overpower ΔT is:

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{1}{(1+\tau_3 S)} \leq \Delta T_0 \{K_4 - K_5 \frac{(\tau_7 S)}{(1+\tau_7 S)} \frac{1}{(1+\tau_6 S)} T - K_6 [\qquad\qquad] - f_2(\Delta I)\}$$

K_4 (nominal)	=	1.116 Technical Specification value
K_4 (max)	=	[ ]$^{+a,c}$
K_5	=	0.020 Technical Specification value
K_6	=	0.00175 Technical Specification value
vessel ΔT	=	56.5 °F smallest ΔT based on evaluation of temperature data

PMA conversions:

ΔT Burndown (PMA_5) = [
Power Cal. (PMA_7) =]$^{+a,c}$

T_{avg} conversion = [
T_{avg} ($RMTE_{Tavg}$) =]$^{+a,c}$

Total Allowance = [ ]$^{+a,c}$ = 4.0 %

Table 10
Seabrook Station Stretch Power Uprate RPS/ESFAS Parameters

Protection Function	Safety Analysis Limit	Nominal Trip Setpoint	TA	CSA	Margin	TS Allowable Value
					+a,c	
Overtemperature Delta T Reactor Trip	1.34	1.21				See Note 1
Overpower Delta T Reactor Trip	1.176	1.116				See Note 2
Steam Generator Water Level - Low Low Reactor Trip	0% span	20% span				≥19.5% span
Steam Generator Water Level - High-High Feedwater Isolation	97.7% span (3)	90.8% span				≤91.3% span

Notes:

(1) Note that 0.5% of ΔT span is applicable to the OTΔT input channels ΔT, T_{avg} and Pressurizer Pressure; 0.25% of ΔT span is applicable to ΔI.

(2) Note that 0.5% of ΔT span is applicable to the OPΔT input channels ΔT and T_{avg}.

(3) Based on Maximum Reliable Indicated Limit (MRIL).

Attachment RAI 4-1

APPLICATION OF THE SETPOINT METHODOLOGY

1.0 Uncertainty Calculation Basic Assumptions/Premises

The equations noted in the Tables contained in Attachment RAI 2-1 are based on several premises. These are:

1) The instrument technicians make reasonable attempts to achieve the Nominal Trip Setpoint as an "as left" condition at the start of each process rack's surveillance interval.
2) The process rack drift will be evaluated for drift magnitude over multiple surveillance intervals.
3) The process rack calibration accuracy will be evaluated over multiple surveillance intervals.

Please note for (1) above that it is not necessary for the instrument technician to recalibrate a device or channel if the "as left" condition is not exactly at the nominal condition but is within the plus or minus of nominal "as left" procedural tolerance. As noted above, the uncertainty calculations assume that the "as left" tolerance (conservative and non-conservative direction) is satisfied on a reasonable, statistical basis, not that the nominal condition is satisfied exactly. This evaluation assumes that the Rack Calibration Accuracy (RCA) and Rack Drift (RD) parameters values noted in the Tables of Attachment RAI 2-1 are satisfied on at least a 95 % probability / 95 % confidence level basis. It is therefore necessary for periodic reverification of the continued validity of these assumptions. This prevents the institution of non-conservative biases due to a procedural basis without the plant staff's knowledge and appropriate treatment.

In summary, a process rack channel is considered to be "calibrated" when the two-sided "as left" calibration procedural tolerance is satisfied. An instrument technician may determine to recalibrate if near the extremes of the "as left" procedural tolerance, but it is not required. Recalibration is explicitly required any time the "as found" condition of the device or channel is outside of the "as left" procedural tolerance. A device or channel may not be left outside the "as left" tolerance without declaring the channel "inoperable" and appropriate action taken. Thus an "as left" tolerance may be considered as an outer limit for the purposes of calibration and instrument uncertainty calculations.

2.0 Process Rack Operability Determination Program and Criteria

As a result of the review of a sample of plant procedures, the equations noted in the Tables of Attachment RAI 2-1 are different from those used in previous Westinghouse uncertainty calculations. One aspect of the equations easily noted is the significance of the calibration process, i.e., it is treated as statistically independent of the drift determination. Another aspect is that if drift and calibration are independent processes, then the determination of equipment operability is changed, i.e., it is not the arithmetic sum of the two uncertainties. The parameter of most interest as a first pass operability criterion is drift ("as found" – "as left") found to be within RD, where RD is the 95/95 drift value assumed for that channel. However, this would require the instrument technician to record both the "as left" and "as found" conditions and perform a calculation in the field. This field calculation has been determined to be impracticable at this time since it would require having the "as left" value for that device at the time of drift determination and thus becomes a records availability/control problem. An alternative for the process racks is the use of a fixed magnitude, two-sided "as found" tolerance about the nominal trip setpoint. It would be reasonable for this "as found" tolerance to be RMTE + RD, where RD is the actual statistically determined 95/95 drift value and RMTE is defined in the Seabrook procedures. However, comparison of this value with the "as left" tolerance utilized in the plant procedures and the Westinghouse uncertainty calculations would yield a value where the "as found" tolerance is less than the "as left" tolerance. This is due to RD being defined as a relative drift magnitude as opposed to an absolute drift magnitude and the process racks being very stable, i.e., no significant drift. Thus, it is not reasonable to use this criterion as an "as found" tolerance in an absolute sense, as it conflicts with the second criterion for operability determination. That is, a channel could be left near zero, found outside the absolute drift criterion, yet be inside the calibration criterion and not exceeding the relative drift criterion. Therefore, a more reasonable approach for the plant staff was determined. The "as found" criterion based on absolute magnitude is the same as the "as left" criterion, i.e., the allowed deviation from the Nominal Trip Setpoint (NTS) on an absolute indication basis is plus or minus the "as left" tolerance. A process loop found inside the "as left" tolerance on an indicated basis is considered to be operable. A channel found outside the "as left" tolerance is evaluated and recalibrated. If the channel can be returned to within the "as left" tolerance, the channel is considered to be operable. This criterion can then be incorporated into plant, function specific calibration and drift procedures as the defined "as found" tolerance about the NTS. At a later date, once the "as found" data is compiled, the relative drift ("as found" – "as left") can be calculated and compared against the RD value. This comparison can then be utilized to ensure consistency with the assumptions of the uncertainty calculations documented in the Tables of Attachment RAI 2-1. A channel found to exceed this criterion multiple times should trigger a more comprehensive evaluation of the operability of the channel.

The proposed Seabrook Station systematic program of drift and calibration review for the process racks is acceptable as a set of first pass criteria. More elaborate evaluation and monitoring may be included, as necessary, if the drift is found to be excessive or the channel is found difficult to calibrate. Based on the above, the total process rack program proposed for Seabrook Station will provide a more comprehensive evaluation of operability than a simple determination of an acceptable "as found".

3.0 Application to the Plant Technical Specifications

The drift operability criteria for the process racks in Section 2 would be based on a statistical evaluation of the performance of the installed hardware. Thus this criterion would change if the Measurement and Test Equipment is changed, or the procedures used in the surveillance process are changed significantly and particularly if the process rack modules themselves are changed, e.g., from pure analog to a mixture of analog and ASIC (Application Specific Integrated Circuit) modules. Therefore, the operability criteria are not expected to be static. In fact they are expected to change as the characteristics of the equipment change. This does not imply that the criteria can increase due to increasingly poor performance of the equipment over time. But rather just the opposite. As new and better equipment and processes are instituted, the operability criteria magnitudes would be expected to decrease to reflect the increased capabilities of the replacement equipment. For example, if the plant purchased some form of equipment that allowed the determination of relative drift in the field, the rack operability would then be based on the RD value.

Sections 1.0 and 2.0 are basically consistent with the recommendations of the Westinghouse paper presented at the June 1994, ISA/EPRI conference in Orlando, FL[1]. Therefore, consistent with the paper, Westinghouse recommends revision of Specifications 3.3-1, "Reactor Trip System Instrumentation Limiting Condition for Operation", Specification 3.3.2, "Engineered Safety Features Actuation System Instrumentation – Limiting Condition for Operation", Table 3.3-1 "Reactor Trip System Instrumentation" and Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation". Table 10 of Attachment RAI 2-1 provides the recommended Nominal Trip Setpoint for each RTS/ESFAS protection function, which was utilized in the Westinghouse uncertainty calculations and determined to be acceptable for use. Table 10 also notes the Westinghouse recommended allowable value for each RTS/ESFAS protection function process rack channel. These recommendations are specific to each input for multiple input functions and are placed in the plant procedures and maintained under plant administrative control. In addition, the plant operability determination processes described in Sections 1.0 and 2.0 are consistent with the basic intent of the ISA paper [2] and the bases sections for the two specifications noted above.

4.0 Determination of Allowable Value

The Allowable Values (AVs) for the Seabrook Technical Specifications are determined by adding (or subtracting) the calibration accuracy of the device tested during the Channel Operational Test to the NTS in the non-conservative direction (i.e., toward or closer to the SAL) for the application. For those channels that provide trip actuation via a bistable in the process racks, the calibration accuracy is defined by the Rack Calibration Accuracy term. The magnitude of the calibration accuracy term is as specified in the station procedures.

An example of the AV calculation is as follows:

- *Steam Generator Level - Low-Low LONF*

  NTS = 20.0 % span
  SAL = 0 % span
  RCA = 0.5 % span
  SPAN = 100 % Level

  AV = NTS - RCA

  AV = 20.0 % - 0.5 %

  AV = 19.5 % span

5.0 References/Standards

[1] Tuley, C. R., Williams, T. P., "The Allowable Value in the Westinghouse Setpoint Methodology – Fact or Fiction?" presented at the Thirty-Seventh Power Instrumentation Symposium (4th Annual ISA/EPRI Joint Controls and Automation Conference), Orlando, FL, June, 1994.

[2] Ibid

ATTACHMENT RAI 42-1
NSAL-04-01

Nuclear Safety Advisory Letter

Westinghouse Electric Company

This is a notification of a recently identified potential safety issue pertaining to basic components supplied by Westinghouse. This information is being provided so that you can conduct a review of this issue to determine if any action is required.

P.O. Box 355, Pittsburgh, PA 15230

Subject: **Hot Leg Switchover Time Clarification**	Number: **NSAL-04-1**
Basic Component: Emergency Core Cooling System Analysis	Date: 01/22/2004
Affected Plants: See attached list in Table 1.	
Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a)	Yes ☐ No ☒
Transfer of Information Pursuant to 10 CFR 21.21(b)	Yes ☐
Advisory Information Pursuant to 10 CFR 21.21(d)(2)	Yes ☐

References:
1. ERG Direct Work Number DW-02-003, transmitted via response letter ERG-03-007, 2/27/2003.
2. Westinghouse Nuclear Safety Advisory Letter NSAL-92-010, "Hot Leg Switchover Methodology", January 1993.
3. Westinghouse Nuclear Safety Advisory Letter NSAL-95-001, "Minimum Cold Leg Recirculation Flow", January 1995.

SUMMARY

Implementation of the hot leg switchover (HLSO) time for realignment to hot leg recirculation after a LOCA may not be consistent with the assumptions used in design basis calculations. Issues regarding the timing and interpretation of the HLSO time involve: whether to begin the realignment process at HLSO time or to complete the realignment process by the HLSO time, minimum acceptable times for completion of the process, and limitations on early HLSO start. This NSAL reinforces discussion in a recent ERG response letter (Reference 1).

Westinghouse was recently made aware of a plant with a designated HLSO time 2.5 hours shorter than the HLSO time calculated in the plant's analysis of record. While this condition was subsequently shown to be acceptable, it may not be so in all cases.

Recommendations are provided to clarify the application of the HLSO time with regard to plant operating procedures and the early initiation of HLSO. Westinghouse has determined that this issue does not represent a substantial safety hazard. Therefore, this issue is not reportable pursuant to the requirements of 10 CFR 21.

Additional information, if required, may be obtained from the originators Telephone (412) 374-4419 / 4912

Originator:(s)	Approved:
B. F. Maurer	J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing	Regulatory Compliance and Plant Licensing
D. J. Fink	
LOCA Integrated Services II	

ISSUE DESCRIPTION

Implementation of the time specified for realignment to hot leg recirculation after a LOCA may not be consistent with design basis calculations. The issues concern the timing and interpretation of the FSAR/EOP designated HLSO time, as they relate to the LOCA analyses that support the designated time, and are summarized as follows:

- Realizing that the hot leg realignment process requires a certain amount of time, should the EOPs and associated operator training be structured to <u>begin the realignment process at</u> HLSO time, or to <u>complete the hot leg realignment process by</u> the HLSO time?
- If the realignment process is to begin at the designated HLSO time, is there a minimum acceptable time to complete the realignment?
- If the realignment process is to be completed at the designated HLSO time, are there limitations regarding how early the realignment process can begin? This issue surfaced recently when Westinghouse was made aware of a plant with a designated HLSO time 2.5 hours earlier than the HLSO time calculated in the plant's analysis of record. While this condition was subsequently shown to be acceptable for the plant in question, it may not be so in all cases.

Issues such as those listed above were recently addressed in an ERG response letter (Reference 1).

TECHNICAL EVALUATION

There are two distinct and separate criteria for establishing an appropriate HLSO time: 1) early enough to preclude the potential for boric acid precipitation in the core and 2) not so early that the hot leg recirculation flow may be inadequate to remove decay heat. Some plants have greatly reduced safety injection flows after realignment to hot leg recirculation when limiting active and passive failures are considered.

The purpose of HLSO is to preclude the potential for boric acid precipitation in the reactor vessel as a result of a cold leg break LOCA. At HLSO time, there must be adequate recirculation flow to provide a core flushing flow for either a hot leg or a cold leg break. Considering solely the potential for boric acid precipitation, HLSO can be viewed as a maximum time, implying that an earlier HLSO is acceptable. However, this is not always the case since core cooling must also be considered.

Because recirculation flows in hot leg recirculation alignment may be reduced over those of cold leg recirculation flows, recirculation flows after HLSO must be verified to be acceptable for core cooling. Acceptability in this case means adequate flows to maintain the vessel inventory for both hot and cold leg breaks. Generic criteria for minimum hot leg and cold leg recirculation flow after HLSO have previously been provided in Reference 2. It is important to note that earlier HLSO increases the hot leg recirculation flow requirements since the flow requirements are based on core boiloff, which will be greater with an earlier HLSO time.

Westinghouse typically establishes the maximum HLSO time in post-LOCA calculations based on boric acid precipitation potential and then performs adequate flow verification at that HLSO time. It is the expectation that EOPs will be written to instruct operators to initiate HLSO at the specified HLSO time, and it is recognized that the HLSO realignment process requires a finite amount of time. The acceptability of this approach is based on the nature of the HLSO calculations and the conservatism in the Westinghouse methodology used to calculate HLSO time. Most significant is the 4% uncertainty margin applied to the boric acid saturation limit of 27.53 weight percent (at atmospheric pressure). This 4% reduction in the boric acid saturation limit typically translates to a margin of more than 2 hours between the recommended HLSO time and the time at which boric acid precipitation may potentially occur.

SAFETY SIGNIFICANCE

For the reasons cited above, the lack of specificity regarding HLSO timing is not believed to be safety significant. With regard to early realignment to HLSO, there are two aspects to consider: the potential for boric acid precipitation, and adequate core cooling. Early switchover would not increase the risk of boric acid precipitation since the hot leg recirculation alignment would still provide adequate core flushing flow at the calculated HLSO time. Delays in switchover (resulting from the time it takes to complete the realignment process) are not safety significant because of the conservatism in the methodology exemplified by the inclusion of the 4% uncertainty margin applied to the boric acid solubility limit. For core cooling however, early switchover would invalidate the HL (hot leg) and CL (cold leg) flow confirmations typically performed by Westinghouse, as discussed in References 2 and 3. Nevertheless, it is likely that the combined HL and CL flow would be adequate to provide core cooling given realistic assumptions for decay heat, SI (safety injection) subcooling, and SI pump performance. Therefore, it is concluded that, if left uncorrected, the issues discussed herein do not represent a substantial safety hazard and consequently are not reportable to the NRC pursuant to the requirements of 10 CFR Part 21.

NRC AWARENESS

The NRC has not been formally notified of this issue.

RECOMMENDED ACTIONS

Westinghouse recommends the following regarding the interpretation of the FSAR/EOP-designated HLSO time. Note that these recommendations are consistent with the discussion in the recent ERG response letter (Reference 1).

1. Westinghouse recommends that FSAR/EOP-designated HLSO times be interpreted as the beginning of the hot leg recirculation realignment process. This is consistent with the definition of ERG footnote (V.01) as amended by Reference 1. For the reasons described above, there is sufficient margin in the HLSO calculations such that the realignment can be completed before the potential for boric acid precipitation exists.

2. If the EOPs are written to begin the hot leg recirculation realignment process prior to the HLSO time (in order to complete the realignment process by the designated HLSO time), Westinghouse recommends that the early initiation be limited to that period of time supported by operations and training experience.

3. If the EOPs are written to allow the hot leg recirculation realignment process to begin well in advance of the HLSO time (such that the realignment process is completed long before HLSO time), licensees should review the post-HLSO recirculation flows with regard to guidance provided in References 2 and 3. This reinforces the revised footnote (V.01) definition in Reference 1 that states "there may be a minimum time requirement prior to aligning for hot leg recirculation".

Table 1

PLANT APPLICABILITY

Plants for which Westinghouse maintains the HLSO analysis of record	Plants which may utilize Westinghouse HLSO methodology
D. C. Cook Units 1 & 2	Angra Unit 1
J. M. Farley Units 1 & 2	Doel Units 1, 2, & 4
Byron Units 1 & 2	Sizewell B
Braidwood Units 1 & 2	Ringhals Units 2, 3, & 4
V. C. Summer	Tihange Units 1 & 3
H. B. Robinson Unit 2	Mihama Units 1 & 2
Shearon Harris Unit 1	Ohi Units 1 & 2
Beaver Valley Units 1 & 2	Takahama Units 1 & 2
Turkey Point Units 3 & 4	
Vogtle Units 1 & 2	
Indian Point Units 2 & 3	
Seabrook Unit 1	
Millstone Unit 3	
Diablo Canyon Units 1 & 2	
Salem Units 1 & 2	
Wolf Creek	
Callaway	
South Texas Units 1 & 2	
Sequoyah Units 1 & 2	
Watts Bar Unit 1	
Kori Units 2, 3 & 4	
Yonggwang Units 1 & 2	
Maanshan Units 1 & 2	
Fort Calhoun	
Koeberg Units 1 & 2	
Almaraz Units 1 & 2	
Vandellos Unit 2	
Ascó Units 1 & 2	
Krsko	
Zorita	

General notes:

1. Two loop plants with UPI (upper plenum injection) do not realign to hot leg recirculation and thus are not affected.
2. The methodology used by Westinghouse for plants in the CE fleet (other than for Fort Calhoun) establishes an acceptability window and is not affected by this issue.

Enclosure 3 to Letter No. SBK-L-04072
Application For Withholding Proprietary Information From Public Disclosure

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-04-1896

September 24, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Reponses to NRC Requests for Additional Information (RAIs) for Seabrook Station Stretch Power Uprate (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1896 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by FPL Energy.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1896, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: W. Macon, NRC
E. Peyton, NRC

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 24th day
of September, 2004

Sharon L. Fiori
Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked Westinghouse Reponses to NRC Requests for Additional Information (RAIs) for Seabrook Station Stretch Power Uprate (Proprietary), dated September 2004 being transmitted by the FPL Energy letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Seabrook Station stretch power uprate is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of plant power uprating.

This information is part of that which will enable Westinghouse to:

(a) Provide information in support of plant power uprate licensing submittals.

(b) Provide plant specific calculations.

(c) Provide licensing documentation support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation associated with power uprate licensing submittals

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations, evaluations, analysis, and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Enclosure 4 to Letter No. SBK-L-04072
Response To Request for Additional Information
for LAR 04-03, Application for Stretch Power Uprate

FPL Energy Seabrook Commitments to
Response to Request for Additional Information Regarding
License Amendment Request 04-03, Application for Stretch Power Uprate

Commitment

FPL Energy Seabrook commits to evaluate the results of the following EPRI MRP programs and to factor them into reactor vessel internals inspections as appropriate

- Material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.
- Evaluation of the effects of irradiation, which include IASCC, swelling, and stress relaxation in pressurized water reactors.
- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on reactor vessel internals.
- Development of a long-term reactor vessel internals aging management strategy