June 1, 2009

NRC 2009-0058
10 CFR 50.90

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Response to Request for Additional Information
License Amendment Request 241
Alternative Source Term

References: (1) FPL Energy Point Beach, LLC letter to NRC, dated December 8, 2008, License Amendment Request 241, Alternative Source Term (ML083450683)

(2) NRC letter to FPL Energy Point Beach, LLC, dated May 13, 2009, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Reactor Systems Branch Related to License Amendment Request No. 241 Alternate Source Term (TAC Nos. ME0219 and ME0220) (ML091110584)

NextEra Energy Point Beach, LLC (NextEra) submitted Point Beach Nuclear Plant (PBNP), Units 1 and 2 proposed License Amendment Request 241 and subsequent supplemental information for Commission review and approval pursuant to 10 CFR 50.90 (Reference 1). The license amendment would revise the current licensing basis to implement the alternative source term (AST) through reanalysis of the radiological consequences of the PBNP Final Safety Analysis Report Chapter 14 Accidents.

Via a letter dated May 13, 2009 (Reference 2), the NRC staff determined that additional information was required to enable the staff's review of the amendment request.

Enclosure 1 provides the NextEra proprietary response to the NRC staff's request.

Enclosure 2 provides the NextEra non-proprietary response to the NRC staff's request.

Enclosure 3 provides Westinghouse authorization letter CAW-09-2579 with accompanying affidavit, Proprietary Information Notice and Copyright Notice. As the enclosure contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit 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 Section 2.390 of the Commission's regulations.

It is requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-09-2579 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.

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

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on June 1, 2009.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 2

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 241
ALTERNATIVE SOURCE TERM

Via a letter dated May 13, 2009 (Reference 1), the NRC staff determined that additional information was required to enable the staff's review of the License Amendment Request (LAR) 241, Alternative Source Term. The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to NRC staff's request.

Question 1

The licensee proposed to use RAVE methodology documented in WCAP-16259-P-A to determine rods in departure from nucleate boiling (DNB) for the locked rotor (LR) event. The Nuclear Regulatory Commission (NRC) safety evaluation report (SER) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052340326), approving WCAP-16259 indicated that the basis of NRC's acceptance of the WCAP is, in part, that "Westinghouse will maintain training guidelines that assure only qualified analysts perform and verify the analyses being performed." Discuss qualifications of the analysts and address how the analysts meet the Westinghouse training guidelines for use of the RAVE methodology documented in the WCAP.

NextEra Response

The RAVE methodology was implemented in accordance with the Westinghouse Quality Management System (QMS), which has been reviewed and approved by the NRC. The analysts and verifiers have been trained and are qualified to perform and verify the RAVE analyses per Westinghouse QMS.

Question 2

The Westinghouse RAVE methodology contains three Westinghouse computer codes, SPNOVA, VIPRE, and RETRAN. Identify and provide the nodalization diagrams for use of SPNOVA, VIPRE, and RETRAN that deviated from those used for reference plants documented in applicable WCAP reports, and justify the deviations.

NextEra Response

The SPNOVA, RETRAN and VIPRE models used in the Point Beach Nuclear Plant (PBNP) locked rotor rods-in-DNB analysis utilize consistent nodalization with the sample 3-loop plant models shown in the RAVE topical report (WCAP-16259-P-A). The VIPRE model for DNBR calculations is the same as that in topical report WCAP-14565-P-A.

The RETRAN methodology topical report (WCAP-14882-P-A) includes a detailed discussion of 2-, 3-, and 4-loop Westinghouse-designed plant models. These models have been reviewed and approved by the NRC. As part of the RAVE methodology Safety Evaluation (SE), the use of these 2-, 3-, and 4-loop Westinghouse-designed plant RETRAN models in the RAVE analyses are accepted, noting the difference in the core nodalization which is increased from **[** **]**^{a,c} used in the RETRAN analyses to **[** **]**^{a,c} used in the RAVE analyses. Nodalization of a sample 3-loop plant reactor pressure vessel which includes **[** **]**^{a,c} in the core is provided in Section C of the RAVE topical report (WCAP-16259-P-A).

The RETRAN reactor pressure vessel model (RPV) for PBNP used in the RAVE analyses is consistent with the plant's RETRAN model (2-loop Westinghouse-designed plant model from WCAP-14882-P-A). The core nodalization of the RETRAN model used in the RAVE analyses is provided in Figure A below. This model uses **[** **]**^{a,c} in the core consistent with the sample 3-loop plant RPV model provided in the RAVE topical report.

An additional nodalization change to the RETRAN model provided in WCAP-14882-P-A which has been added for all RETRAN analyses (including the RAVE analyses) was made for the nodalization of the reactor coolant system hot legs. Since the approval of WCAP-14882-P-A, the hot leg modeling was enhanced to address temperature measurement interactions for pressurizer insurge and outsurge. This hot leg model enhancement, which has been applied in other RETRAN analyses performed by Westinghouse, consisted of dividing each hot leg control volume into three equal control volumes. Although it was needed only for the hot leg connected to the pressurizer, all loops were divided in the same manner.

A one-to-one mapping is used for the SPNOVA and VIPRE whole core model nodalization consistent with the WCAP-16259-P-A. The active core is modeled with [ ]$^{a, c}$. Each assembly in the core is modeled with [ ]$^{a, c}$. This corresponds to [ ]$^{a, c}$. The individual code models are consistent with the models described in WCAP-12394-A for SPNOVA and WCAP-16259-P-A for the VIPRE whole core model used in the core response calculations. The PBNP VIPRE whole core model radial nodalization for the core (feedback) response calculation is illustrated in Figure B. This nodalization is consistent with the sample 3-loop whole core model nodalization provided in Section C of WCAP-16259-P-A. The only difference is that PBNP is a 2-loop plant and the core contains 121 assemblies (compared to 157 assemblies in the 3-loop core).

The nodalization of the VIPRE subchannel model used in the DNBR calculations is the same as the model shown in Figure 4-2 of WCAP-14565-P-A.

a, c

Figure A. Reactor Pressure Vessel Nodalization for PBNP, Units 1 and 2
(2-loop Westinghouse-designed plant)

XXX	Volume Number
XXX	Junction Number

a, c

Figure B. VIPRE Whole Core Nodalization for PBNP, Units 1 and 2
(2-loop Westinghouse-designed plant)

Question 3

Section 6.2 of Enclosure 3 to the licensee's letter dated December 8, 2008, stated that in the analysis of the steam generator tube rupture (SGTR), "The equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR, at which time the operators have completed the actions necessary to terminate the steam release from the ruptured SG. Pressure between the ruptured SG and the primary system is such that the ruptured SG is not overfilled." The consequences of a SGTR depend largely on the ability of the operator to take necessary actions to terminate the primary-to-secondary break flow. The licensee did not indicate what is the operator action time from the start of the event assumed in the analysis to terminate break flow. If the break flow continues for an extended period of time, the secondary side of the SG may be filled and water may enter the steam line, which results in unanalyzed conditions. As a result of the January 1982 SGTR event at the Ginna plant, NRC questioned the assumptions used in the Ginna SGTR analysis, which assumed that the event is terminated in 30 minutes. In response to the NRC staff concerns, a subgroup of utilities in the Westinghouse Owner's Group was formed to address the licensing issues associated with a SGTR event on a generic basis. The subgroup submitted and NRC approved a topical report, WCAP-10698-P-A, "SGTR Analysis methodology to Determine the Margin to Steam Generator Overfill."

Discuss the SGTR mitigation strategy credited in the analysis and provide the basis for the mitigation strategy. Provide a sequence of the event listing the operator actions credited in the SGTR analysis and justify adequacy of the assumed operator actions and associated times. Also, discuss the computer code used to determine the margin to SG overfill and show it is an NRC-approved code. In addition, discuss whether the methodology documented in WCAP-10698 is applicable and needed to apply to the Point Beach plant for the SG overfill prevention or not. The Point Beach plant and Ginna plant are Westinghouse two-loop plants with similar rated thermal power levels. Among other Westinghouse plants, Ginna performed and NRC approved the SGTR reanalysis using the WCAP-10698 methodology. If the licensee determines that the WCAP methodology is not applicable and a reanalysis of the SGTR event is not needed, provide rationale for the determination. If the licensee determines that the reanalysis using the WCAP methodology is needed, it should provide the results of the SGTR reanalysis to the NRC staff for review.

NextEra Response

Consistent with plants of the same vintage, the current SGTR licensing basis for PBNP Units 1 and 2 consists of a simplified thermal-hydraulic analysis to determine the mass of primary-to-secondary break flow and the mass of steam released to the atmosphere for input to a radiological consequences analysis. This simplified thermal-hydraulic analysis assumes that primary-to-secondary break flow continues until 30 minutes from the start of the event and includes conservative assumptions to provide appropriate radiological dose consequences. This simplified analysis does not specifically examine margin to steam generator overfill. The current licensing basis for PBNP Units 1 and 2 is consistent with plants which received their operating license prior to the Ginna SGTR event which prompted development of the WCAP-10698-P-A steam generator tube rupture methodology.

Following the approval of the WCAP-10698-P-A methodology, the NRC did not require plants that received their operating licenses prior to the Ginna SGTR event to update their analyses to the new methodology (the exception was Ginna, who had to apply the new methodology). The current licensing basis SGTR analysis for PBNP Units 1 and 2 does not consider whether or not break flow termination can be demonstrated in 30 minutes and does not address the issue of steam generator overfill following an SGTR. However, in order to confirm the conservative nature of the radiological doses calculated using the 30-minute licensing basis SGTR analysis presented in LAR 241(Reference 2), and to specifically examine margin to steam generator overfill, supplemental SGTR analyses were performed for PBNP Units 1 and 2 at EPU conditions. These supplemental analyses model operator responses leading to break flow termination consistent with PBNP emergency operating procedure EOP-3. PBNP-specific operator action times were used as shown in Table 1.

The supplemental SGTR analyses used a selective implementation of the modeling provided in the WCAP-10698-P-A methodology. For PBNP the supplemental SGTR analyses did not include consideration of the limiting single failure (consistent with the current licensing basis SGTR analysis for PBNP), analyzed nominal plant conditions without uncertainties, and did not apply increased conservatism on the initial secondary mass assumption. The supplemental analyses were performed using the LOFTTR2 computer code which was approved by the NRC along with the WCAP-10698-P-A methodology. The analyses were performed for PBNP Units 1 and 2 in a manner consistent with those accepted in the safety evaluation report (SER) for D.C. Cook (Reference 3). Additionally, the impact of NSAL-07-11 (Reference 4) was incorporated into the margin to overfill analysis. Each supplemental analysis applied a limiting set of operating conditions to provide conservative results for the analysis being performed (i.e., the supplemental dose analysis used assumptions that maximized ruptured steam generator steam releases, and the supplemental margin to overfill analysis used assumptions that maximized ruptured steam generator water volume).

The supplemental SGTR dose analysis modeled 0% tube plugging and 577°F RCS average temperature with the Unit 2 Model Δ47 steam generator (bounding for Unit 1). The evaluation included consideration of a maximum safety injection flow rate and a minimum auxiliary feedwater flow rate. The calculated sequence of events is presented in Table 2 which shows break flow termination in approximately 53 minutes. Figures 1 through 6 contain plant transient responses to the tube rupture event including primary and secondary pressure, primary-to-secondary break flow, primary-to-secondary flashing fraction, and secondary steam releases. Table 3 compares the dose consequences calculated using the results of the 30-minute licensing basis analysis as presented in LAR 241 to the dose consequences calculated using the results of the supplemental SGTR dose analysis. The doses were calculated using identical assumptions with the only difference being the mass transfers used as input. The resulting doses for the supplemental SGTR dose analysis with break flow termination at 53 minutes are lower than those presented in LAR 241. Thus, the supplemental SGTR dose analysis confirms that the dose analysis presented in LAR 241 is bounding.

The limiting margin to overfill calculation models 10% tube plugging, and 558°F RCS average temperature with the Unit 1 Model 44F steam generator (bounding for Unit 2) and nominal decay heat. The sensitivity to decay heat documented in WCAP-10698-P-A was revisited for the PBNP margin to overfill evaluation to address NSAL-07-11 (Reference 4) and it was determined that nominal 1971 American Nuclear Society (ANS) decay heat is more limiting than 1971+20% ANS decay heat for the PBNP margin to overfill analysis. The evaluation includes consideration of maximum safety injection and auxiliary feedwater (AFW) flow rates. The calculated sequence of events is presented in Table 4 which shows break flow termination in approximately 44 minutes. Figures 7 and 8 contain the plant transient pressure response and the primary-to-secondary break flow response to the tube rupture event. Figure 9 presents the ruptured steam generator water volume as compared to the available steam generator water volume demonstrating that steam generator overfill does not occur.

The results of the supplemental SGTR analyses demonstrate that the 30-minute licensing basis analysis presented in LAR 241 is bounding from a radiological dose consequences perspective, and that margin to overfill is maintained for a SGTR at PBNP Units 1 and 2.

Table 1: Operator Action Times Modeled for Supplemental Evaluations

Operator action time to isolate the ruptured steam generator	6 minutes or LOFTTR2 calculated time to reach 41% narrow range level in the ruptured steam generator, whichever is longer
Operator action time to initiate cooldown following isolation of the ruptured steam generator	17 minutes
Plant response to complete cooldown	LOFTTR2-calculated
Operator action time to initiate depressurization following completion of cooldown	3 minutes
Plant response to complete depressurization	LOFTTR2-calcualted
Operator action time to terminate Emergency Core Cooling System (ECCS) flow	2 minutes
Break flow termination time resulting from primary and secondary pressure equalization	LOFTTR2-calculated

Table 2: Sequence of Events for the Supplemental Dose Analysis

Event	Time (sec)
SGTR occurs	0
Reactor trip	77
Initiation of safety injection	202
Initiation of AFW	377
Isolation of ruptured steam generator	804
Initiation of cooldown with intact steam generator	1822
Break flow flashing stops	1968
Termination of cooldown	2442
Initiation of depressurization	2622
Termination of depressurization	2690
Termination of safety injection	2811
Termination of break flow	3144

Table 3: Comparison of Radiological Consequences

Scenario	Location (time interval)	LAR 241 Analysis (rem TEDE)	Supplemental Analysis (rem TEDE)
Pre-Accident Iodine Spike	EAB (0-2 hours)	2.0	0.7
	LPZ (0-30 hours)	0.2	0.05
	CR (0-30 days)	1.9	0.8
Accident-Initiated Iodine Spike	EAB (0-2 hours)	0.6	0.3
	LPZ (0-30 hours)	0.1	0.02
	CR (0-30 days)	0.5	0.3

Table 3 Notes:
EAB – Exclusion Area Boundary
LPZ – Low Population Zone
CR – Control Room

Table 4: Sequence of Events for the Supplemental Margin to Overfill Analysis

Event	Time (sec)
SGTR occurs	0
Reactor trip	64
Initiation of AFW	64
Initiation of safety injection	163
Isolation of ruptured steam generator	360
Initiation of cooldown with intact steam generator	1380
Termination of cooldown	1854
Initiation of depressurization	2034
Termination of depressurization	2096
Termination of safety injection	2217
Termination of break flow	2632

Figure 1: Supplemental SGTR Dose Analysis
RCS and Secondary Pressures

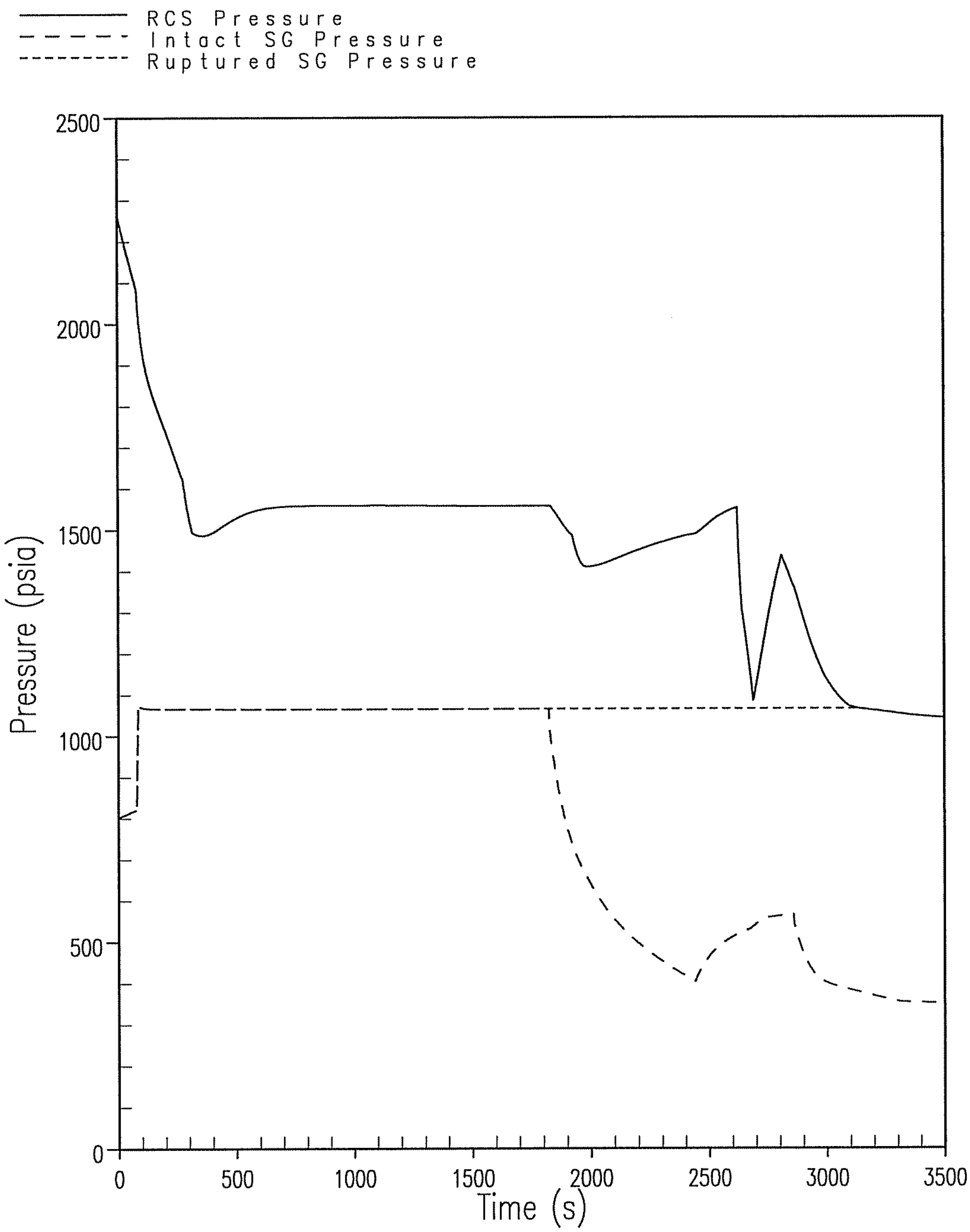


Figure 2: Supplemental SGTR Dose Analysis
Primary-to-Secondary Break Flow Rate

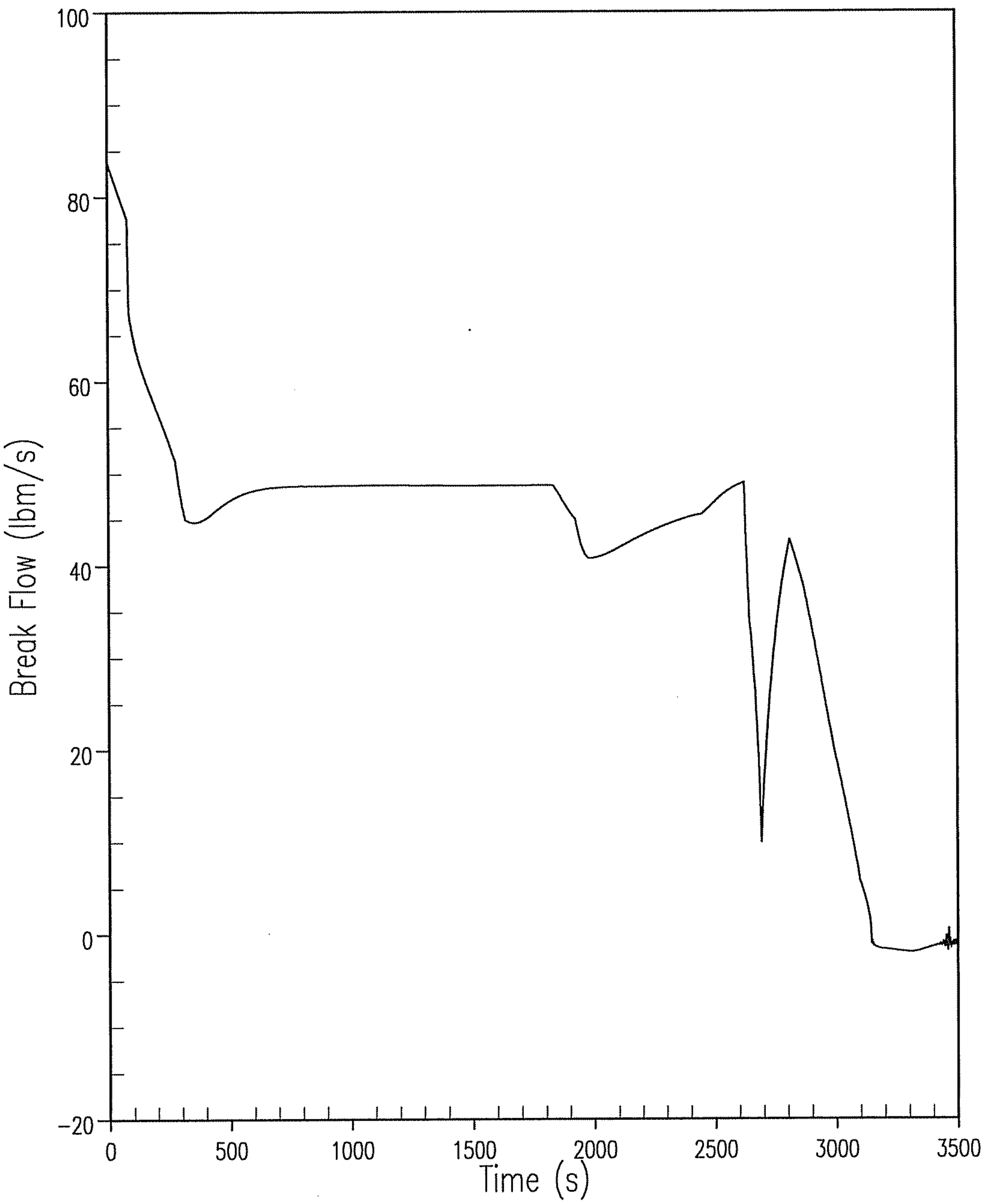


Figure 3: Supplemental SGTR Dose Analysis
Integrated Primary-to-Secondary Break Flow

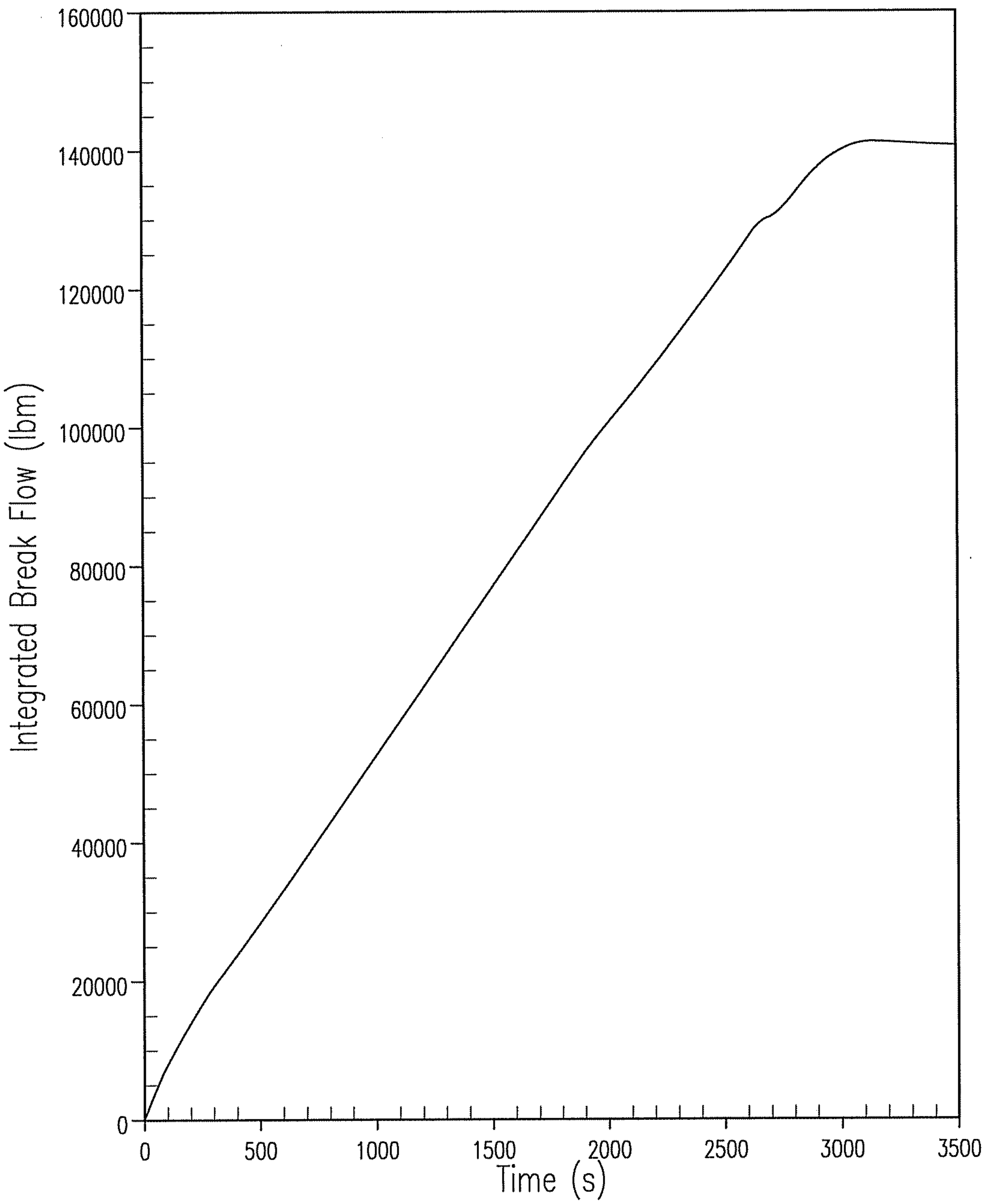


**Figure 4: Supplemental SGTR Dose Analysis
Primary-to-Secondary Break Flow Flashing Fraction**

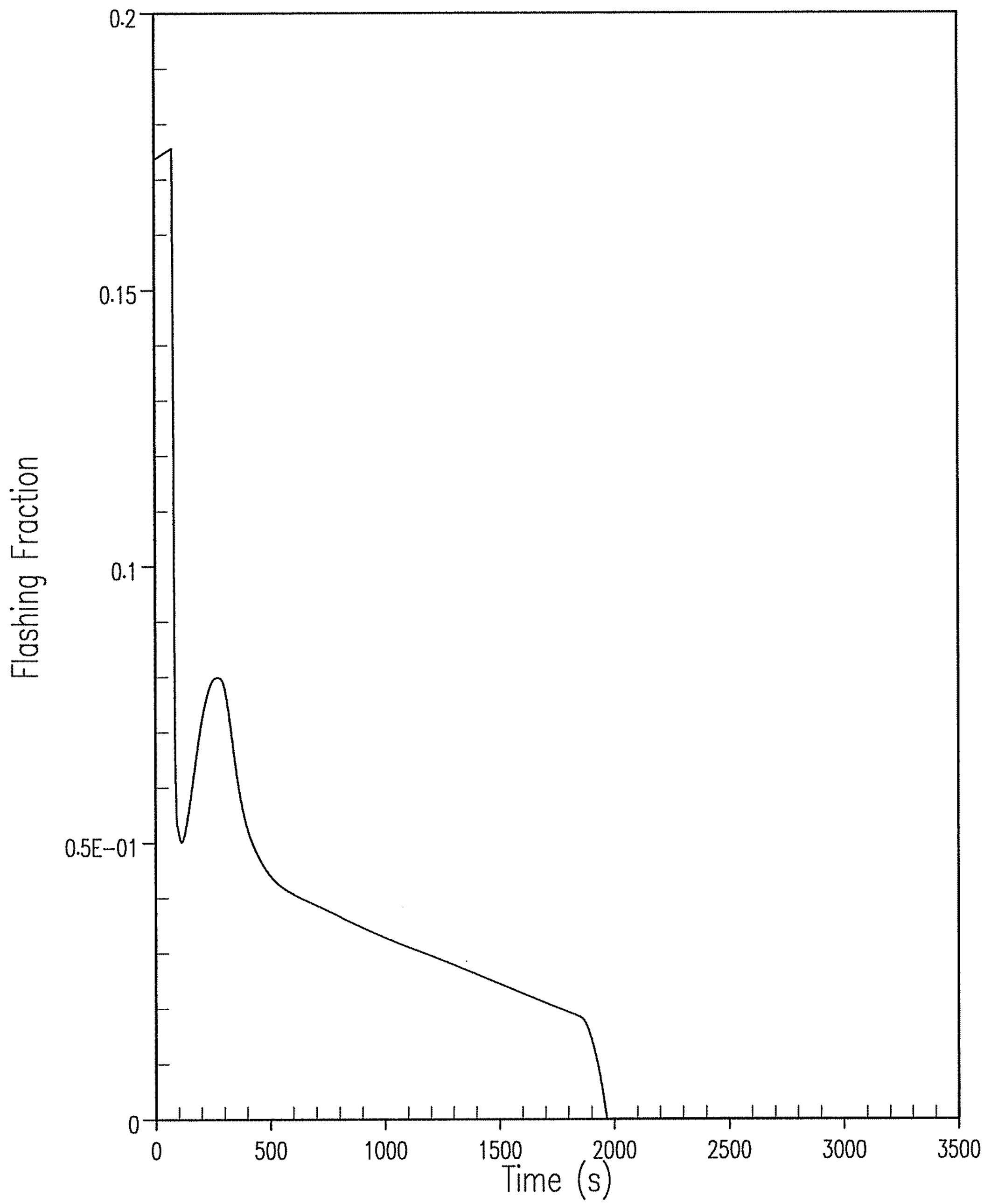


**Figure 5: Supplemental SGTR Dose Analysis
Integrated Primary-to-Secondary Flashed Break Flow**

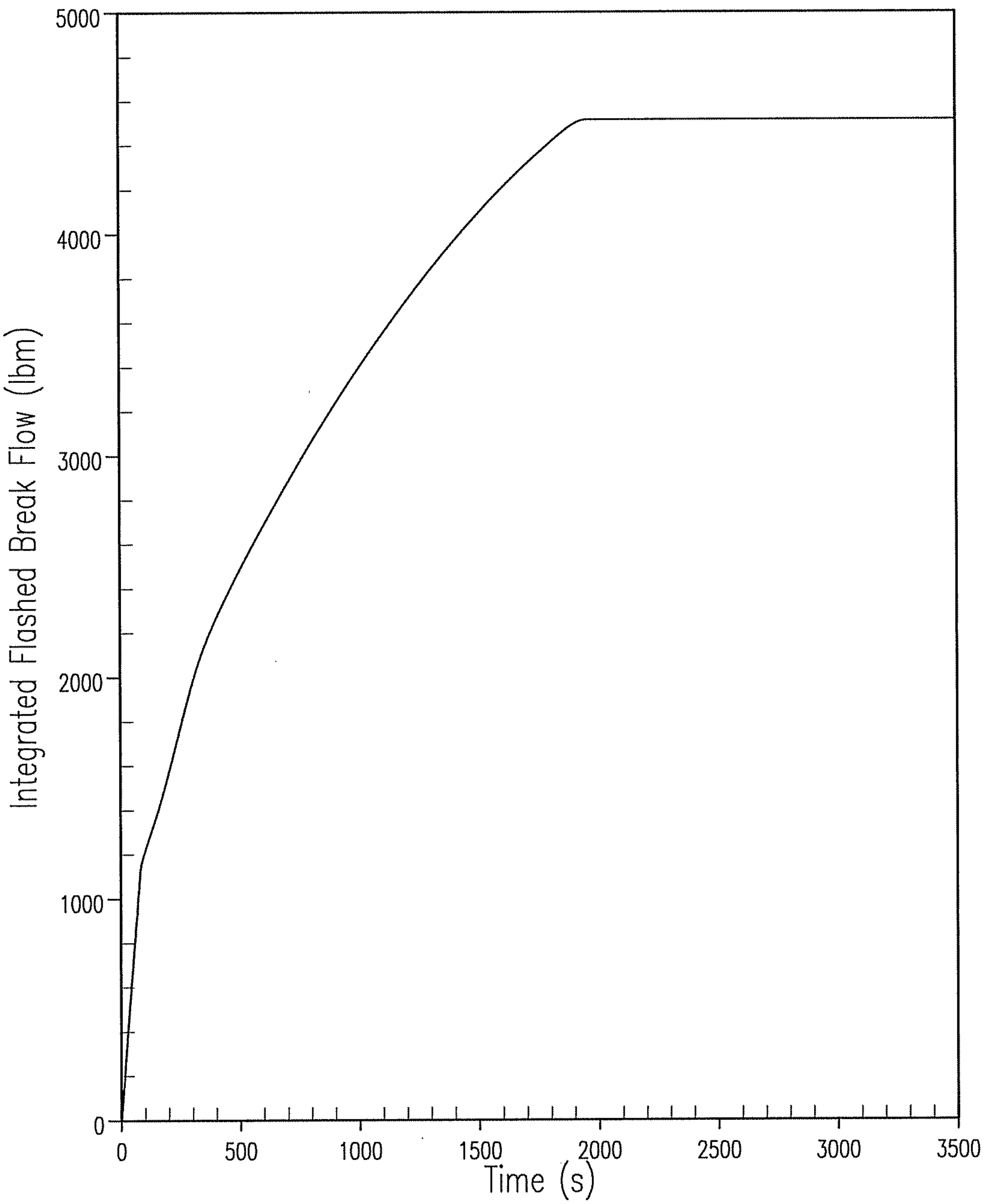


Figure 6: Supplemental SGTR Dose Analysis Secondary Mass Release Rates

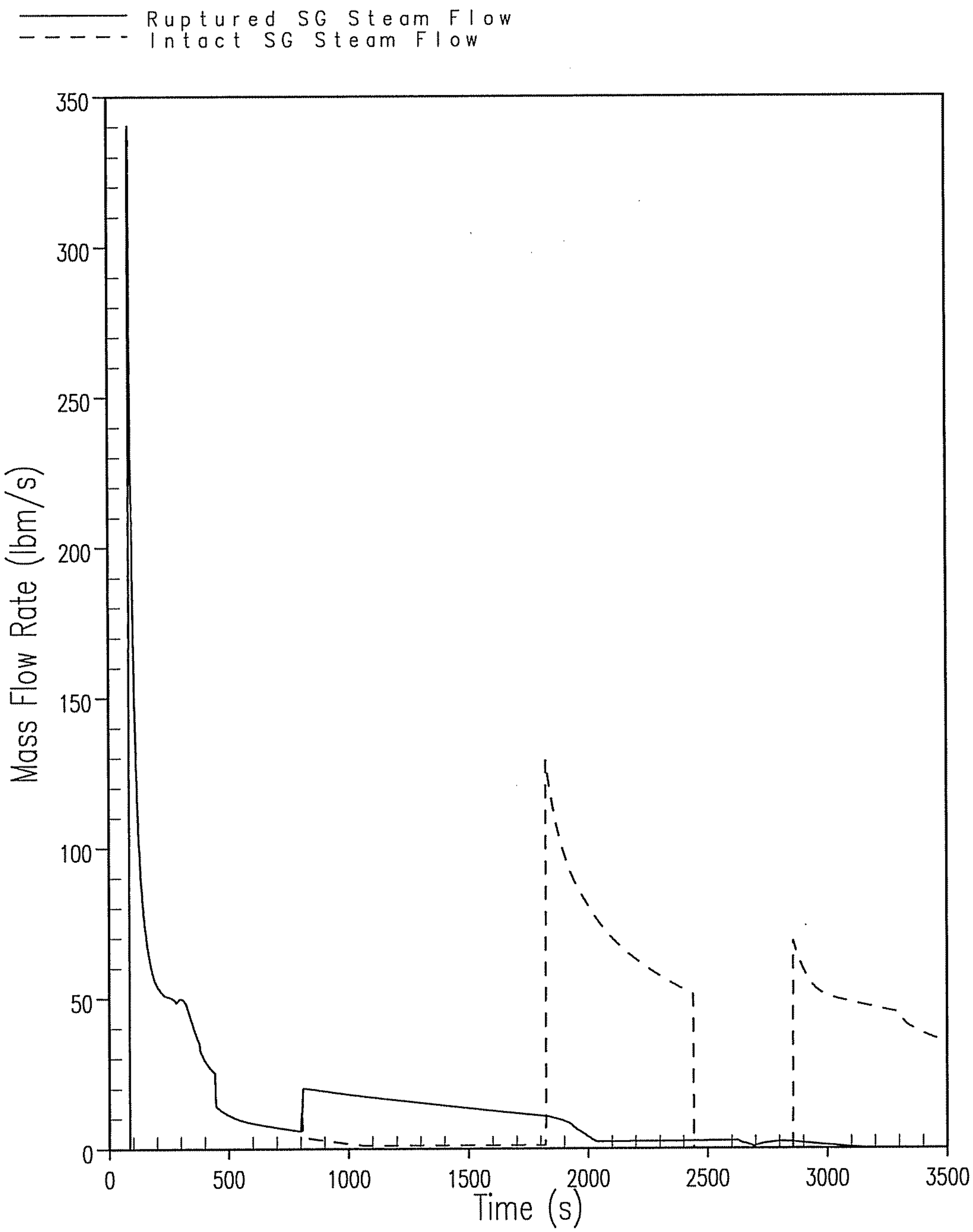


Figure 7: Supplemental SGTR Margin to Overfill Analysis
RCS and Secondary Pressures

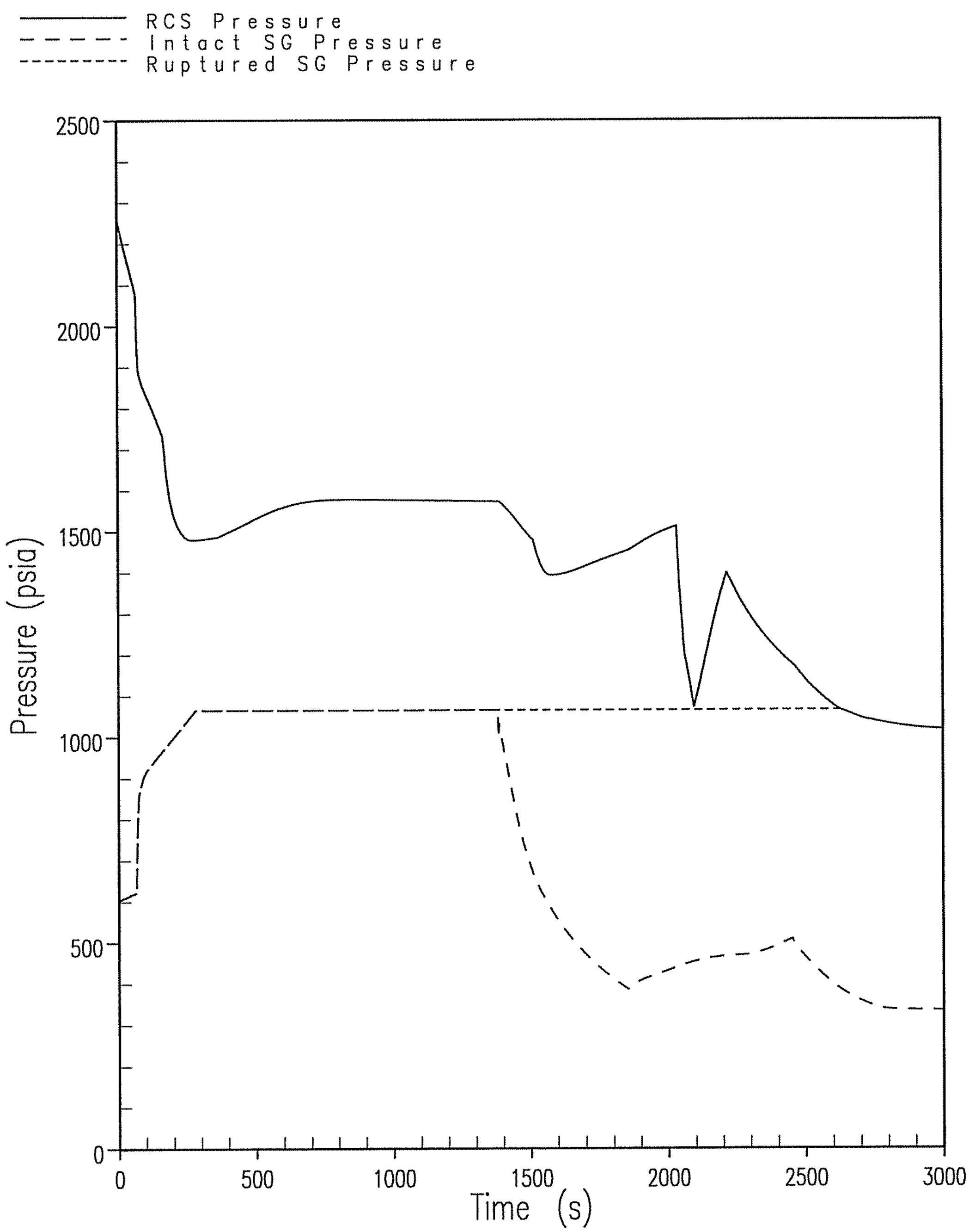


Figure 8: Supplemental SGTR Margin to Overfill Analysis Primary-to-Secondary Break Flow Rate

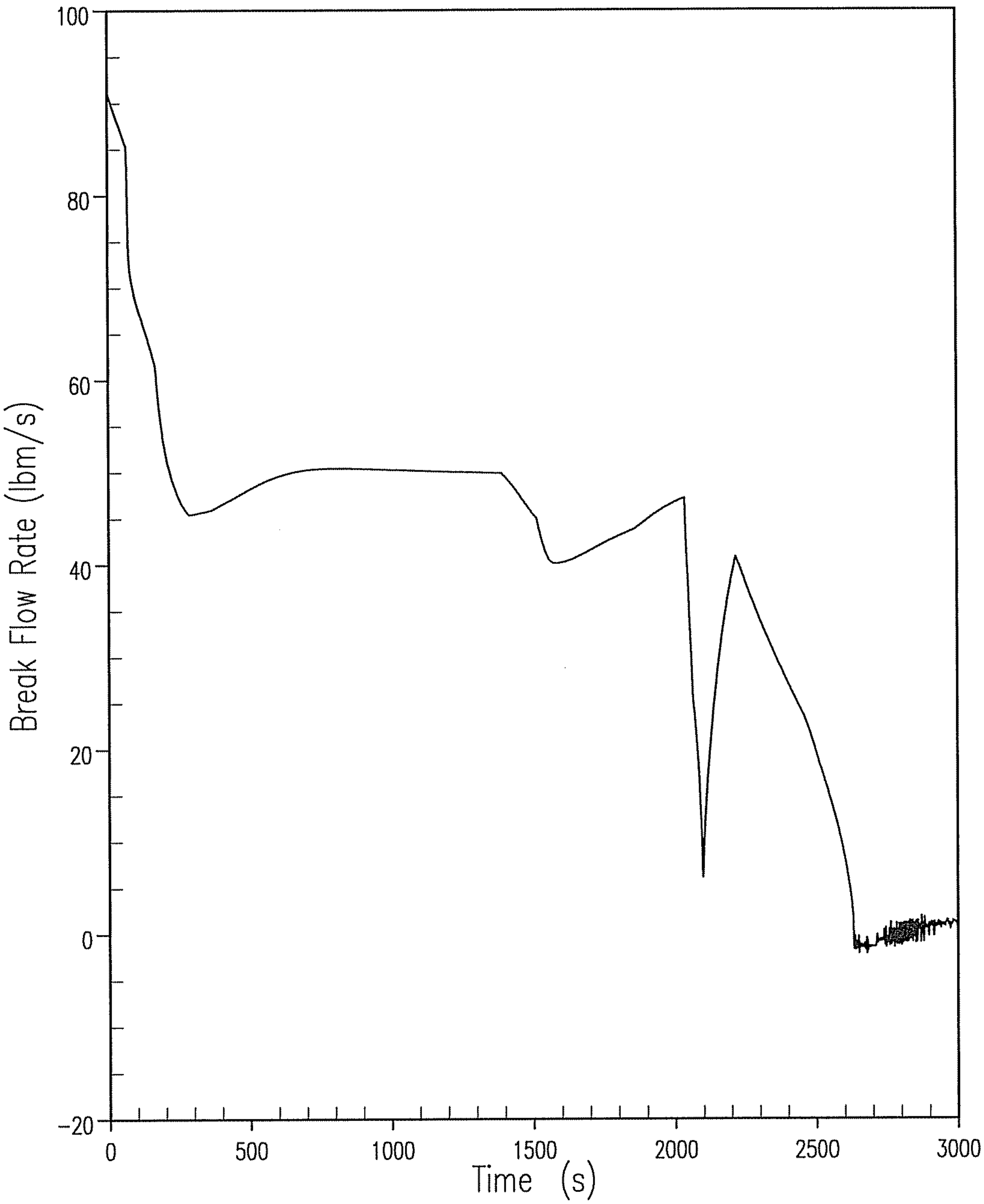
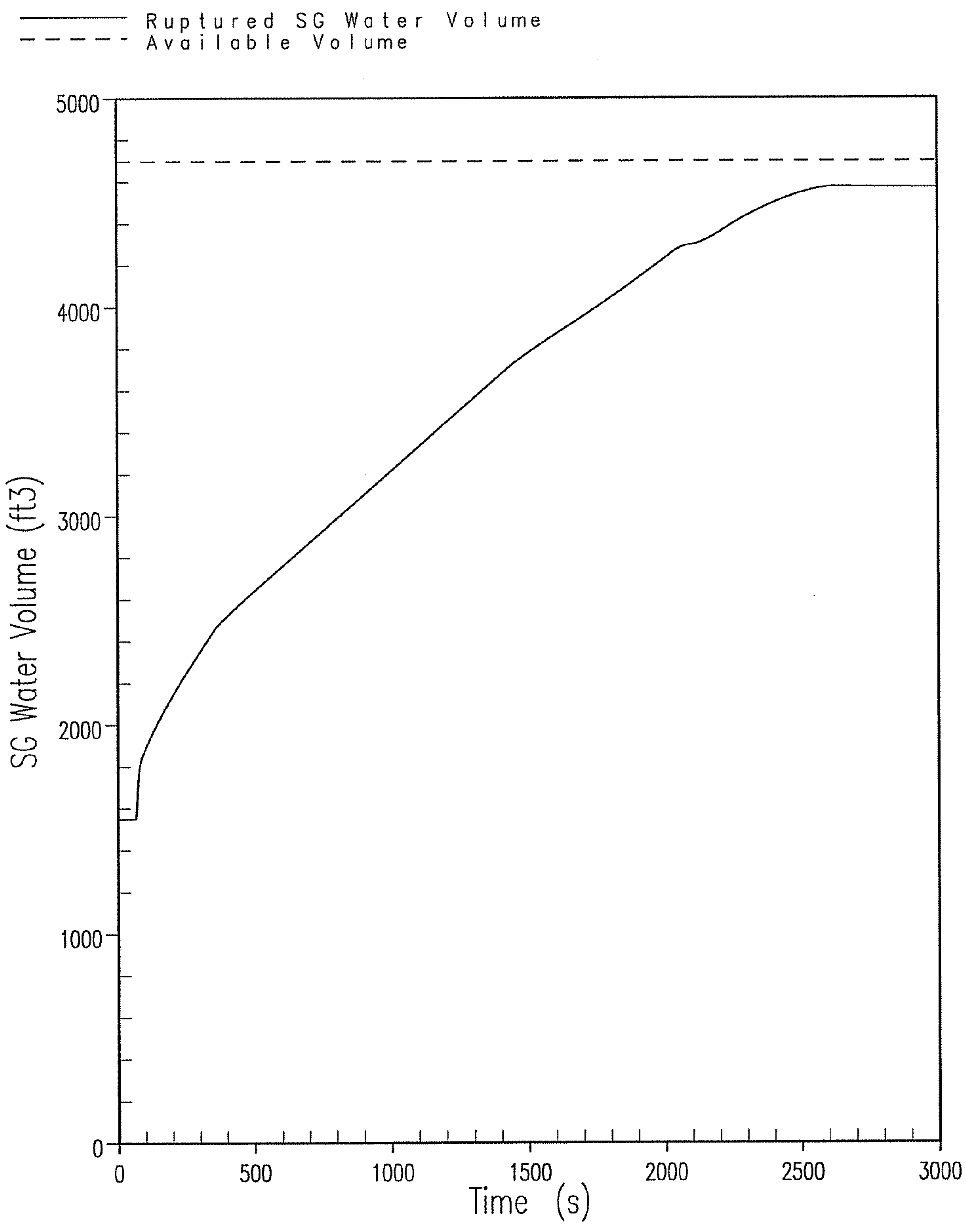


Figure 9: Supplemental SGTR Margin to Overfill Analysis
Ruptured SG Water Volume



Question 4

In response (Enclosure 7 to letter of December 8,2008) to condition 1 on the use of RETRAN, the licensee indicated that RETRAN will be used in the analysis not only for the locked rotor event, but also for the following events: (1) excessive increase in steam flow; (2) steam line break; (3) loss of external electrical load; (4) loss of all alternating current power to the station auxiliaries; (5) loss of normal feedwater flow; (6) loss of reactor coolant flow; and (7) uncontrolled rod withdrawal at power.

Although RETRAN was approved by NRC on a generic basis, the licensee should provide a discussion to address the adequacy of the specific plant application of RETRAN for analysis of the events identified above as event (1) through (7) by showing that: the analysts using RETRAN to perform the analysis are adequately qualified; the values of input parameters appropriately represent the plant conditions or reflect limiting core operating conditions when applicable; the results of thermal-hydraulic and system responses for the analysis are within the approved applicable ranges of RETRAN; and there is no mathematical unstable conditions. (The same requests are applied to the use of VIPRE for licensing applications).

Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," outlines a process that a licensee can use to remove cycle-specific parameters from the plant-specific Technical Specifications (TS) to a licensee-controlled document entitled, "Core Operating Limit Report" (COLR). A necessary element of that process is that a licensee includes specific methodologies in TSs. In accordance with the GL guidance, justify that the topical reports that documented the approved RETRAN and VIPRE codes are not referenced in TS 5.6.4, "Core Operating Limits Report (COLR)".

In addressing compliance of condition 2 on the use of VIPRE, the licensee indicated that "continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272P-A. Following the GL guidance, justify that the topical report, WCAP-9272-A, is not referenced in TS 5.6.4.

NextEra Response

WCAP-16259-P-A, which utilizes the RETRAN and VIPRE codes, was only used to confirm the assumption that 30% of rods in DNB used in the locked rotor radiological dose consequence analysis was conservative. RETRAN was not utilized in the locked rotor radiological dose consequence analysis.

The discussion of the use of RETRAN for events (1) through (7) identified above in this LAR was provided for information only. The Limitations, Restrictions, and Conditions associated with the plant application of RETRAN for PBNP Units 1 and 2 are discussed on Pages A-6 and A-7 in Appendix A of Attachment 5 in License Amendment Request 261, Extended Power Uprate, dated April 7, 2009 (ML091250569). The Limitations, Restrictions, and Conditions associated with the plant application of VIPRE for PBNP Units 1 and 2 are discussed on Pages A-9 and A-10 in Appendix A of Attachment 5.

As discussed above, WCAP-16259-P-A, which utilizes the RETRAN and VIPRE codes was only utilized to confirm the assumption that 30% of rods in DNB used in the locked rotor radiological dose consequence analysis was conservative. Additionally, the locked rotor radiological dose consequence analysis is not associated with any COLR parameter. Therefore, consistent with the guidance in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits from Technical Specifications, dated October 4, 1988 (ML031150407), the RETRAN and VIPRE codes do not need to be listed as references in Technical Specification (TS) 5.6.4, "Core Operating Limits Report (COLR)," since they were not used to determine any core operating limits contained in the COLR.

WCAP-9272-P-A is referenced in TS 5.6.4.b.(2).

Question 5

Section 6.3 of Enclosure 3 to the licensee's letter dated December 8, 2008, stated that "30% of the fuel rods in the core are assumed to suffer damage due to DNB" in the LR radiological analysis.

Provide a basis for the assumption of 30 percent fuel failure due to DNB used in the LR radiological analysis. Discuss the methodology for determination of the number of failed fuel rods due to DNB during a LR event. Justify that the core with assumed 30 percent fuel failure maintains in a coolable geometry during a LR event.

NextEra Response

The 30% fuel failure (rods-in-DNB) assumption used in the locked rotor radiological dose consequence analysis is based on preliminary evaluations of plant-specific radiological and locked rotor accident analyses. The 30% rods-in-DNB limit has been shown to be acceptable for the plant dose calculations, while maintaining sufficient margin in the safety analysis in support of future reloads.

The RAVE locked rotor rods-in-DNB analysis (similar to the current point kinetics analysis) is performed in two steps: 1) core response analysis and 2) hot rod analysis. As discussed in WCAP-16259-P-A, the separation of DNBR calculation from the reactivity feedback calculation allows separate conservatisms to be applied to different models. The locked rotor rods-in-DNB analysis is performed using the same Revised Thermal Design Procedure (WCAP-11397-P-A) that has been applied to the current PBNP DNB analyses. Therefore, the initial conditions (reactor power, pressurizer pressure and RCS temperature) are assumed to be at their nominal values and the uncertainties are included in the DNBR limit. All key parameters (i.e., Moderator Temperature Coefficient (MTC), trips reactivity, Doppler feedback, etc.) are conservatively modeled to maximize the core power response during the transient. A sensitivity study is performed to determine the conservative directions for the key parameters and is included as part of the RAVE analysis. The PBNP specific sensitivity study for the locked rotor rods-in-DNB analysis is included in Enclosure 6 of Reference 2.

The method for determining rods-in-DNB percentage remains unchanged from the current point kinetics methodology. Any fuel rod having a high power factor that results in DNBR lower than the limit during the locked rotor transient is conservatively assumed to be in DNB. The percentage of rods-in-DNB is then determined based on a rod power factor vs. % of fuel rods in the core census table for the reference core loading plan. The census table is verified for each fuel reload.

The 30% rods-in-DNB assumption was confirmed to be met using the RAVE methodology for the locked rotor analysis performed for the PBNP Units 1 and 2.

Although the fuel rods which experience DNB are conservatively assumed to fail and release their gap activity for the purpose of the locked rotor radiological dose consequence analysis, core coolability limits are met and the coolable geometry is maintained during a locked rotor event.

Question 6

In response (Enclosure 7 to letter of December 8, 2008) to condition 1 on the use of VIPRE, the licensee indicated that the WRB-1 correlation with the associated safety departure from nucleate boiling ratio (DNBR) limit of 1.17 was used in the DNB analysis for the Point Beach 14x14 422V+ fuel. Discuss the design of 14x14 422V+ fuel, compare it with the fuel designs that are acceptable for use of the WRB-1 correlation, and justify that the use of WRB-1 for the 14x14 422V+ fuel is within the applicable range of the WRB correlation with the associated safety DNBR limit of 1.17. If the application of the WRB-1 correlation was previously approved by NRC, provide the author, date and title of the NRC SER approving the WRB-1 correlation for use in the DNBR analysis for the Point Beach 14x14 422V+ fuel.

NextEra Response

The WRB-1 correlation with a 95/95 correlation limit DNBR of 1.17 for Westinghouse 14x14 fuel design is documented in WCAP-8762-P-A, New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids, dated 1984. (The non-proprietary version is WCAP-8763-A.)

The applicability of WRB-1 to Point Beach with the 14x14 422V+ fuel is documented in a letter from the licensee to the NRC, 14x14, 0.422" OD VANTAGE + (422V+) Fuel Design, dated December 16, 1997, following the NRC approved Fuel Criteria Evaluation Process (FCEP) in WCAP-12488-A, Westinghouse Fuel Criteria Evaluation Process, dated 1994.
(The non-proprietary version is WCAP-14204-A.)

Qualification of WRB-1 with the VIPRE code is discussed in WCAP-14565-P-A, VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, dated 1999. (The non-proprietary version is WCAP-15306-NP-A.) The WRB-1 95/95 correlation limit of 1.17 remains unchanged with the VIPRE code.

Question 7

In addressing compliance (Enclosure 7 to letter of December 8, 2008) with condition 5 on the use of RAVE, the licensee indicated that the impact of exceeding 30 percent void fraction limit was investigated and it was determined to be conservative with respect to over pressurization during a LR event. Discuss the impact study of the void fraction on over pressurization and provide the results to support the conclusion that the void fraction greater than 30 percent will result in a higher reactor coolant system pressure during a LR over pressurization event.

NextEra Response

The RAVE analyses (similar to the current point kinetics analyses) are typically performed in two steps: 1) core response analysis and 2) hot rod analysis. The separation of these two parts allows us to implement conservative assumptions which are typically applied in opposite directions [

]a, c.

However, the locked rotor overpressure analyses are performed in one step only without a hot rod calculation.

The 30% void limit is intended to be used with the RAVE analyses using a two step approach so that core response calculations conservatively predict a nuclear power vs. time transient. For the locked rotor rods-in-DNB and peak clad temperature (PCT) analyses, the 30% void limit is met for the core (feedback) response calculations.

Table A summarizes the maximum void fraction results obtained for the sensitivity cases presented for the RAVE locked rotor overpressure analysis (included in Enclosure 6 of Reference 2).

Table A a, c

[

]

As seen from the results presented in Table A, [

]$^{a, c}$. Table B summarizes the additional sensitivity cases performed to test the impact of parameters that can change maximum void fraction reached during the transient.

Table B

a, c

[

]

As seen from the results presented in Table B, [

]$^{a, c}$.

The final locked rotor overpressure analysis was performed combining the most conservative assumptions determined via the sensitivity study. The peak RCS pressure of 2653 psia and the maximum void fraction of 31% were obtained in the final locked rotor overpressure analysis.

Based on the sensitivity results, the RAVE locked rotor overpressure analysis is conservatively performed and exceeding the 30% void limit is acceptable.

Question 8

The NRC SER (from NRC to W. J. Johnson (Westinghouse) in letter dated November 26, 1990) approving the use of SPNOVA imposed the following conditions:

(a) the comparison must include both the advanced nodal code (ANC) calculational uncertainty and the ANCISPNOVA calculated difference in isothermal temperature coefficient (ITC), if the SPNOVA ITC uncertainty is determined by comparison to ANC;

(b) additional benchmarking is required, if SPNOVA is applied to fuel and core designs that differ significantly from those included in the benchmark data discussed in Section 3.2.1 of the SER; and

(c) the uncertainties of transient application of SPNOVA are required to assure an acceptable margin to the fuel safety limits and must be provided in event-specific submittal.

Address the compliance with the above conditions.

NextEra Response

(a) In a separate letter dated March 29, 1996 (from Westinghouse to R. C. Jones (NRC), Process Improvement to the Westinghouse Neutronics Code System, NSD-NRC-96-4679), a process improvement that has resulted in streamlining and consolidating the Westinghouse neutronics code system was discussed. As concluded in that letter, the implementation of the advanced nodal code (ANC) solution method in SPNOVA eliminated the solution differences between ANC and SPNOVA and also eliminated the SPNOVA normalization step to the ANC conditions. Therefore, it is not expected to have differences between the results of the SPNOVA and ANC codes.

As part of the SPNOVA model generation, the SPNOVA isothermal temperature coefficient (ITC) check was performed for the PBNP SPNOVA model. This check is still included in Westinghouse work procedures for historical reasons. As a result of this check, the SPNOVA ITC compared favorably against the ANC results and did not require additional penalties.

The calculational uncertainties (including ITC/MTC) are addressed as part of the RAVE methodology (WCAP-16259-P-A) and were explicitly included in the RAVE analyses.

(b) The PBNP core design did not differ significantly from those included in the benchmark data discussed in SPNOVA SE. The SPNOVA and ANC codes are considered equivalent in terms of their solution method (see Item (a)). [

]$^{a, c}$ All SPNOVA results compared favorably against the ANC results.

(c) The uncertainties used in the RAVE locked rotor analyses are addressed in the RAVE methodology topical (WCAP-16259-P-A). A sensitivity study was performed to determine the conservative directions for the key parameters used in the PBNP locked rotor rods-in-DNB analysis. The maximum uncertainty allowances specific to Point Beach were modeled for the key parameters in the conservative directions.

References

(1) NRC letter to FPL Energy Point Beach, LLC, dated May 13, 2009, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Reactor Systems Branch Related to License Amendment Request No. 241 Alternate Source Term (TAC Nos. ME0219 and ME0220) (ML091110584).

(2) FPL Energy Point Beach, LLC Letter to NRC, dated December 8, 2008, License Amendment Request 241, Alternative Source Term (ML083450683).

(3) D.C. Cook Safety Evaluation, dated October 24, 2001, Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments (TAC Nos. MB0739 and MB0740) (ML012690136).

(4) NSAL-07-11, dated November 15, 2007, Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology.

ENCLOSURE 3

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

WESTINGHOUSE AUTHORIZATION LETTER, ACCOMPANYING AFFIDAVIT, PROPRIETARY INFORMATION NOTICE, AND COPYRIGHT NOTICE
CAW-09-2579

(8 pages follow)

Westinghouse

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Proj letter ref WEP-09-63 P-Attachment
Our ref: CAW-09-2579

May 22, 2009

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Point Beach Nuclear Plant, Units 1 and 2 – Response to Request for Additional Information from Reactor Systems Branch Related to License Amendment Request No. 241 Alternate Source Term (TAC Nos. ME 0219 and ME 0220)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-09-2579 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 Group.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-09-2579, 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: George Bacuta (NRC OWFN 12E-1)

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)
R. Morrison (ECE 4-16A) 1L, 1A
L. Walker (ECE-320H) 1L, 1A
D. Dominicis (ECE-323C) 1L, 1A

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 & Plant Licensing

Sworn to and subscribed before me
this 22nd day of May, 2009

Sharon L. Markle

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance & 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 as "Point Beach Nuclear Plant, Units 1 and 2 – Response to Request for Additional Information from Reactor Systems Branch Related to License Amendment Request No. 241 Alternate Source Term (TAC Nos. ME 0219 and ME 0220)," being transmitted by FPL Group 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 Point Beach Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for alternate source term submittals.

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

(a) Provide input to the Nuclear Regulatory Commission for review of the Point Beach alternate source term submittals.

(b) Provide results of customer specific calculations.

(c) Provide licensing 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 alternate source term submittals.

(b) Westinghouse can sell support and defense of the technology to its customer 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 information 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).

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