FENOC

FirstEnergy Nuclear Operating Company

James H. Lash
Site Vice President

724-682-5234
Fax: 724-643-8069

November 18, 2005
L-05-165

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

Subject: Beaver Valley Power Station, Unit No. 1
BV-1 Docket No. 50-334, License No. DPR-66
Response to a Request for Additional Information (RAI dated September 28, 2005) in Support of License Amendment Request No. 320

On October 4, 2004, FirstEnergy Nuclear Operating Company (FENOC) submitted License Amendment Request (LAR) Nos. 302 and 173 by letter L-04-125 (Reference 1). This submittal requested an Extended Power Uprate (EPU) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 and is known as the EPU LAR.

On April 13, 2005, FENOC submitted LAR 320 for BVPS Unit No. 1 by letter L-05-069 (Reference 2). This submittal requested the Technical Specification changes necessary for operation of BVPS Unit No. 1 with the replacement steam generators and is known as the RSG LAR.

By letter dated September 28, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) pertaining to LAR 320. It is noted that the September 28, 2005 RAI questions are applicable to both the RSG and EPU LARs.

During a conference call on October 13, 2005, the NRC added three additional RAI questions to those contained in the September 28, 2005 RAI, and extended the requested response date to November 18, 2005.

Enclosure 1 contains the Non-Proprietary FENOC responses to the September 28, 2005 RAI questions. The responses to the three additional RAI questions have been included in Enclosure 1 as Questions 15, 16, and 17.

Enclosure 2 contains the complete responses, including the information proprietary to Westinghouse Electric Company LLC, along with 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.

Enclosure 3 contains supplemental information requested during a November 2, 2005 call concerning the BVPS Unit No. 2 Leading Edge Flow Meter. Although unrelated to the September 28, 2005 RAI, the requested supplemental information is provided in this transmittal.

The responses and supplemental information provided by this transmittal have no impact on the proposed Technical Specification changes, or the no significant hazards consideration, transmitted by References 1 or 2.

No new regulatory commitments are contained in this submittal. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Licensing, at 330-315-7243.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 18, 2005.

Sincerely,

James H. Lash

Enclosures:

1. Non-Proprietary Responses to RAI dated September 28, 2005
2. Affidavit and Proprietary Responses to RAI dated September 28, 2005
3. BVPS Unit No. 2 Leading Edge Flow Meter Supplement Information

References:

1. FENOC Letter L-04-125, License Amendment Request 302 and 173, dated October 4, 2004.

2. FENOC Letter L-05-069, License Amendment Request 320, dated April 13, 2005.

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

L-05-165 Enclosure 1

Non-Proprietary Responses to RAI dated September 28, 2005

REQUEST FOR ADDITIONAL INFORMATION (RAI)

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)

REPLACEMENT STEAM GENERATORS (RSGs) LICENSE AMENDMENT REQUEST (LAR)

AND EXTENDED POWER UPRATE (EPU)

By letter dated April 13, 2005, Agencywide Documents Access and Management System (ADAMS) Accession No. ML051080573, FENOC (licensee) proposed changes to the BVPS-1 operating license to allow operation with replacement steam generators (LAR 1A-320). The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's application and determined that it will need the additional information identified below to complete its review.

1. Question

The licensee indicated in its April 13, 2005, RSG LAR application that much of the supporting analyses for the request were contained in the previously submitted extended power uprate LAR dated October 4, 2004. Based upon the licensee's July 8, 2005, response to the BVPS-1 and 2 EPU LAR RAI dated May 5, 2005, it appears that the feedwater line break (FWLB) accident analysis methodology has been significantly altered from the Updated Final Safety Analysis Report (UFSAR) Analysis-of-Record (AOR). The AOR models the break flow as follows: "a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator [SG]." The analysis assumptions and initial conditions are built around this conservative modeling technique. For example, a full double-ended break area is assumed in order to rapidly deplete the affected SG inventory. The EPU analysis attempts to model the dynamic SG liquid level within the affected SG and credits an early transition from liquid to steam discharge. This methodology change results in a cooldown of the reactor coolant system (RCS) similar to that during a main steamline break (MSLB) accident analysis.

The methodology change brings into question the analysis assumptions and initial conditions. For example, crediting steam discharge with a large-break size promotes a more rapid depressurization of the affected SG, which results in an earlier reactor trip and auxiliary feedwater system initiation as well as a more substantial cooldown of the RCS. Furthermore, since the transition from liquid to steam discharge has a first order effect, the ability of the Westinghouse (W) analytical tools to accurately predict dynamic conditions with both the affected and unaffected SGs takes on more importance. The pedigree of these models needs further assessment. The licensee is requested to discuss the apparent changes in methodology and either demonstrate that they conform to previously approved methodologies for FWLB and MSLB accident analyses, or provide justification for the change in methodology.

Additionally, since the FWLB analysis is intended to evaluate more than one acceptance criteria which may be mutually exclusive to each other, the limiting case for one criteria may not be limiting with respect to the other criteria, e.g., bulk boiling versus peak

pressure. Please verify that the FWLB analyses are limiting with respect to each of the acceptance criteria considered.

Response:

The Feedwater Line Break analysis performed in support of the Replacement Steam Generators and Extended Power Uprate License Amendment Requests assume a saturated liquid blowdown (0% quality) from the initiation of the break until approximately the time that the feedring is uncovered. At that point, the effluent quality is transitioned to 35% and subsequently held at 35% until there is no liquid mass and the entire steam generator is only steam. Saturated steam blowdown (100% quality) is then assumed for the remainder of the transient consistent with WCAP-9230 (Reference 1). This modeling conservatively bounds the quality of the feedwater line break effluent as a function of steam generator mass/water level for feedring steam generators based on NOTRUMP calculations for a detailed feedring steam generator model. This quality transition results in a rapid blowdown of the faulted steam generator minimizing both the short term (up to the time of reactor trip) and the long term (following reactor trip) heat removal capability of the faulted steam generator. Furthermore, the heat removal capability of the intact steam generators is also minimized by initiating the event with a minimum mass in the intact steam generators.

These assumptions are consistent with the current licensing basis feedwater line break analyses performed in support of the BVPS submittals. There have been no feedwater line break methodology changes made in the modeling of the break effluent. Since approximately 1981, the BVPS UFSARs have incorrectly stated that "a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator." This error has been entered into the FENOC corrective action process and will be corrected with the next annual update. Major Assumption #6 in Section 14.2.5.2.2 of the BVPS-1 UFSAR and Major Assumption #7 in Section 15.2.8.2 of the BVPS-2 UFSAR will be revised as follows:

"A conservative feedwater line break discharge quality is assumed. This minimizes the heat removal capability of the affected steam generator."

The current discharge quality assumption (described in the first paragraph of this response) was adopted for feedring steam generators in the late 1970s and has been used for plants with a feedring type steam generator since that time. Other examples of plants whose current licensing basis FWLB analysis contain an identical assumption are Diablo Canyon, Salem, Cook 2, Millstone 3, Callaway and Vogtle.

For a MSLB, saturated steam blowdown (100% quality) is assumed for the duration of the transient as this maximizes the cool down of the RCS. No credit is taken for any entrained water in the break effluent as this would deplete the steam generator inventory faster and result in a less limiting RCS cool down.

With respect to overpressurization, the feedwater line break event is bounded by the Loss of Load (LOL)/ Turbine Trip (TT) event, in which assumptions are made to conservatively calculate the RCS and main steam system (MSS) pressure transients. For a feedwater line break event, turbine trip occurs following reactor trip, whereas for the LOL/TT event, the turbine trip is the initiating fault. Thus, the primary to secondary power mismatch and resultant RCS and MSS heat up and pressurization transients are always more severe for the LOL/TT event. For this reason, it is not necessary to calculate the maximum RCS or MSS pressures for the feedwater line break event. Thus, the feedwater line break analysis follows the methodology of WCAP-9230.

2. Question

Due to the Nuclear Regulatory Commission (NRC) staff's concerns and the need to complete its safety evaluation (SE) in a timely manner, the NRC staff is requesting that it perform an audit of the supporting W engineering calculation. Dates of November 7-10, 2005, have been agreed to by your staff to allow the NRC staff to conduct this audit. The issues identified below will need to be resolved in order for the NRC staff to complete its SEs related to the BVPS-1 RSG LAR and BVPS-1 and 2 EPU LAR.

a. **The FWLB analysis methodology presented in the RSG and EPU LARs is significantly different than the methodology described in the BVPS-1 and 2 UFSARs. Specifically, the UFSARs state that the faulted SG break flow characteristics are conservatively modeled as liquid discharge; whereas, the new analysis employs a best-estimate technique which predicts steam discharge prior to reactor trip.**

1. **Identify when and how the methodology changed.**

Response

There has been no change to the methodology. The current UFSAR text is incorrect. See response to Question 1.

2. **Provide a break spectrum analysis to identify the limiting break size.**

Response

Figures 2-1 and 2-2 provide the requested break size spectrum results for the BVPS units. The results show that acceptable analysis results are obtained for all break sizes up to and including a break equivalent to the feedwater inlet nozzle.

Figure 2-1 shows the margin to hot leg saturation (MARHLS) as a function of break size during the post-trip heatup. Note that the upper line on Figure 2-1 credits the lead-lag on low steam pressure to provide earlier steamline isolation and the lower line on Figure 2-1 does not credit the lead-lag. The shape of the curve is dominated by the very conservative density feedback assumptions made in the analysis. With the larger break sizes, the cool down effects and the resultant core criticality and power generation become significant. The very conservative density feedback assumptions are based on pre-trip conditions and do not account for the reduced feedback conditions that would be experienced post-trip with the most reactive control rod stuck out of the core. This results in a premature prediction for return-to-critical conditions and artificially limits the cool down and maximizes the heat addition to the RCS compared to what would actually occur. It is worth noting that more realistic post-trip reactivity feedback assumptions would increase the margin to subcooling by more than 30 degrees, and therefore the cases presented in the submittals remain bounding with respect to sub-cooling margin.

Figure 2-2 shows the time of rod motion (via low-low steam generator water level signal - LLSGL) and the time of steamline isolation – SLI (via low steam pressure signal). For all break sizes, the low-low steam generator water level setpoint is reached before the low steam pressure setpoint.

Figure 2-1

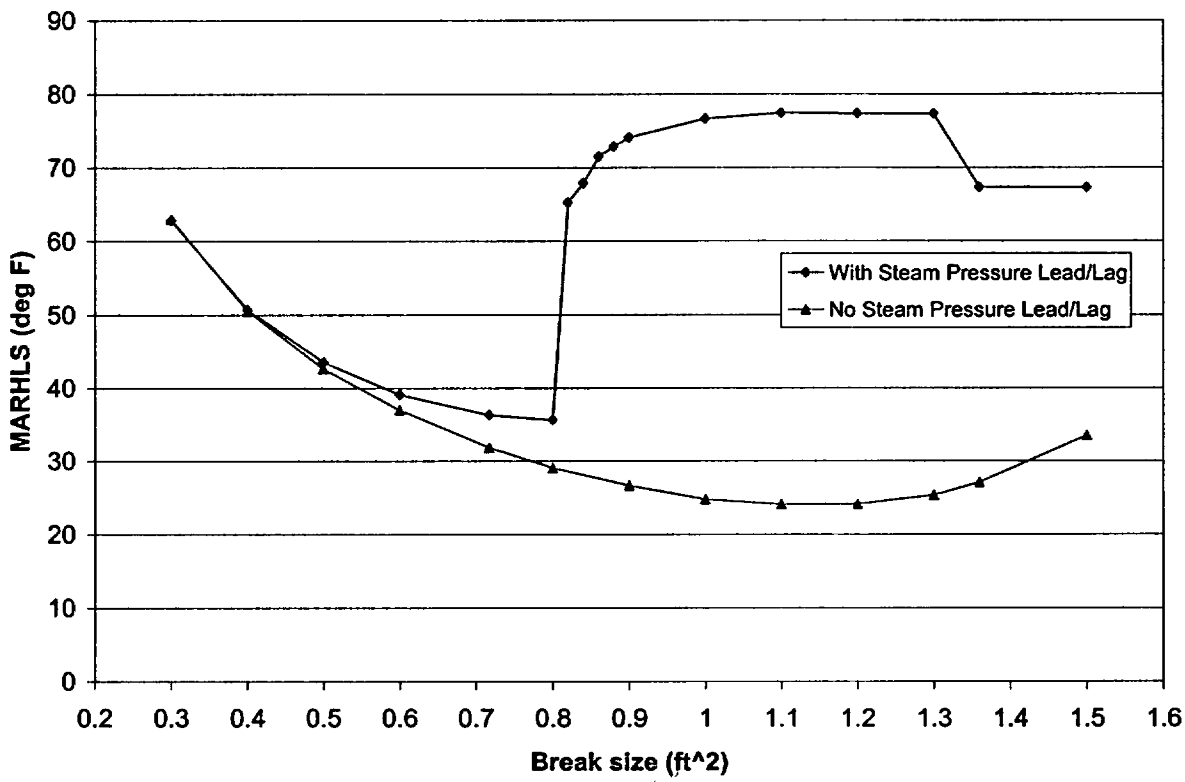
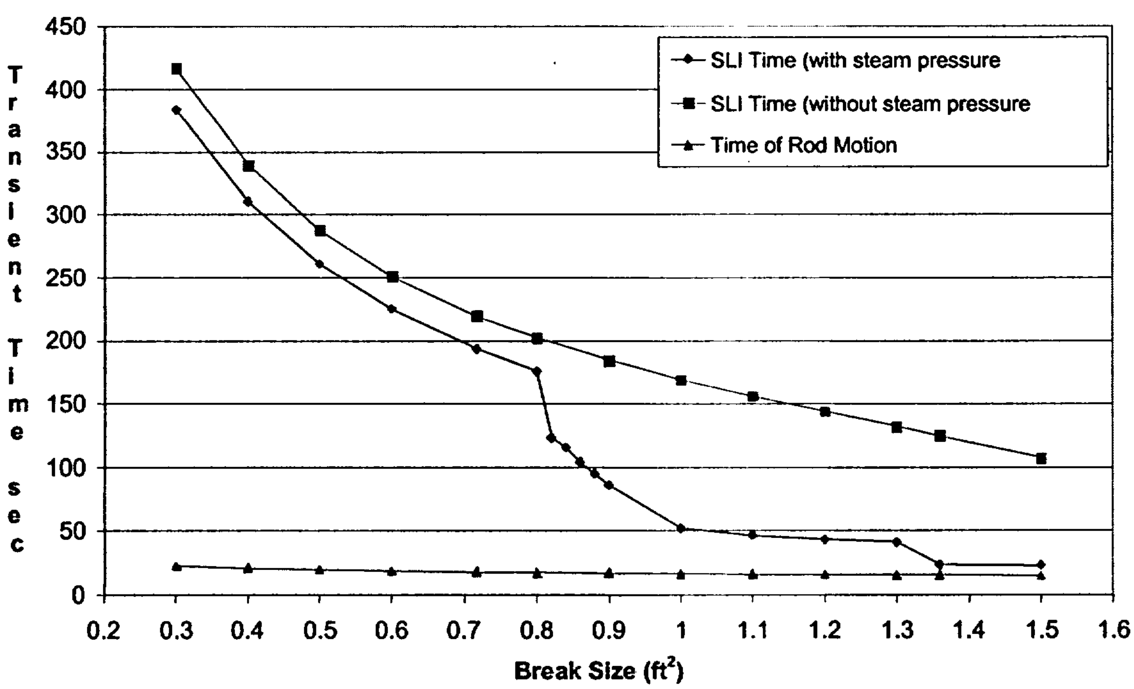


Figure 2-2



Since these results indicate that, contrary to WCAP-9230 statements, the largest possible break size may not yield the most conservative results, an issue has been entered into the Westinghouse Corrective Action Process (CAP) to assess the appropriateness of several assumptions made in the Westinghouse feedwater line break methodology for feedring type steam generators. Break size and the effect of break size on the trip mass are two of the assumptions being investigated. Sensitivities have been run in the past for selected steam generator models, such as the Model D steam generator, which demonstrate that the largest break size is limiting. However, this CAP will determine whether this is an appropriate assumption for all of the feedring steam generators. It will also determine whether the assumed trip mass should vary with break size. The issue will be addressed separate from the BVPS RSG and EPU submittals. The above information demonstrates that the issue is not a concern for BVPS due to available margin in the feedwater line break analysis.

3. **Quantify the degree of uncertainty associated with the "indicated" SG downcomer liquid level and its effects on the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS) response.**

Response

The LOFTRAN code does not model the secondary steam generator water levels directly but rather models the secondary in terms of a total mass of water in the steam generator. The uncertainty associated with the indicated SG level is addressed in a setpoint study for the BVPS units. For normal environmental conditions, the level uncertainty is ± 10% (BVPS-1) and +7%/-10.3% (BVPS-2) of narrow range span (NRS). This "normal" value is used in setting conservative values for the initial SG levels in the analysis, consistent with the methodology of WCAP-9230. For adverse environmental conditions, the level uncertainty is 19.6% of NRS for BVPS-1 and 20.5% NRS for BVPS-2. This "adverse" value is applied in determining the conditions at which the low-low steam generator water level reactor trip and AFW actuations are initiated, consistent with the methodology of WCAP-9230. To account for this adverse environmental condition, the FWLB analyses assume a reactor trip mass corresponding to 0% NRS. Also, the total steam generator mass corresponding to the low-low steam generator water level (0% NRS) is reduced by an additional 10%. This results in a very conservative mass in the steam generator at the time of reactor trip, thereby minimizing the heat removal capability of the secondary system following reactor trip.

4. **Quantify the degree of uncertainty associated with "actual" SG downcomer liquid level and its effects on break discharge characteristics.**

Response

The impact of the uncertainty on downcomer liquid level on break discharge characteristics has not been quantified. However, information on the prediction of the blowdown characteristics is contained in WCAP-9236 (Reference 2). This WCAP supports NOTRUMP's ability to reasonably predict mass blowdown characteristics for a vessel, which inherently includes the "actual" level prediction. WCAP-9236 is Reference 3 in the feedwater line break topical report (WCAP-9230) and was submitted to the NRC at approximately the same time as WCAP-9230. Following the time that WCAP-9230 and WCAP-9236 were submitted, additional NOTRUMP models were developed. Similar to the discussion in WCAP-9230 for the Model D steam generators, bounding blowdown characteristics for feedring steam generator models (e.g., Model F and Model 51) were developed for application to feedwater line break analyses.

The other major impact of the NOTRUMP model is in the timing of the low level trip signal for the narrow range span. The NOTRUMP model includes the prediction of the "indicated" SG level from the delta-pressure between the tap locations of the narrow range span. This calculation addresses the dynamics of the feedwater line break transient.

Also, see response to Question 2 part a.6.

In addition, there are numerous conservative assumptions that are part of the Westinghouse Feedwater Line Break methodology that has been applied for at least 30 years.

For example:

- Use of the maximum pre-trip density coefficient for reactivity feedback provides a significant potential for a return-to-criticality and power generation, especially for the feedring style steam generators which would be expected to have a significant cooldown.
- The faulted loop SG mass is biased high to delay reactor trip and AFW actuation thereby creating a more limiting plant condition at the time of reactor trip.
- The intact loop SG mass is biased low to minimize the mass in the steam generators at time of reactor trip and thereby minimize the post trip heat removal capability of the steam generators.
- The low-low steam generator trip mass is biased low to delay reactor trip and AFW actuation.
- Minimum AFW capacity is assumed with no credit for increases in the AFW due to transient variations in the steam pressure.
- A maximum feedwater line purge volume is assumed before "cold" AFW is delivered to the intact steam generator(s).
- Minimum Safety Injection capacity is assumed at maximum fluid temperature conditions.
- A conservative acceptance criterion of no boiling in the hot leg to ensure that the core remains coolable and geometrically intact.
- No credit is taken for the heat sink provided by the thick metal masses in the steam generator or the RCS.
- A conservative decay heat model is assumed.
- A conservative pump heat value is assumed.
- No credit for charging or letdown is assumed.
- No credit is assumed for steam dumps thereby maximizing the RCS heat up.
- No credit is assumed for any feedwater reaching the intact steam generators prior to or following reactor trip, thereby minimizing the total steam generator masses at the time of reactor trip.

5. Investigate the modeling of the break flow through the downcomer feed ring and out through the break.

Response

This scenario is analyzed for BVPS-1 and BVPS-2 and presented in the Section 5.3-17 of the LAR. The flow area of the feedring is 0.922 ft^2 for a Model 54F steam generator (BVPS-1) and 0.717 ft^2 for a Model 51M steam generator (BVPS-2). The additional case considered for BVPS-2 of 1.36 ft^2 accounts for a failure of the feedring assembly because the feedring assembly is not considered a safety class item nor is it part of the pressure boundary for a Model 51 steam generator.

6. With regard to the use of the "more detailed SG design codes," describe their use and interaction with LOFTRAN and RETRAN codes and discuss the licensing history and approval of these codes.

Response

The RETRAN code is not being used for BVPS-1 or BVPS-2 at this time. Thus, RETRAN is not discussed here.

The following provides a history on the licensing interactions for the LOFTRAN and NOTRUMP codes.

The LOFTRAN code (Reference 3) was submitted by Westinghouse to the U. S. Nuclear Regulatory Commission (NRC) for regulatory review in 1972 via WCAP-7907 (Reference 4).

Summaries of the Westinghouse Feedwater Line Rupture methodology (References 2 and 3) were found acceptable for review by the NRC (Reference 5,) and Reference 5 indicates that, if the subject reports (References 2 and 3) were submitted to the NRC for review "this month, we plan to complete our review in March 1979." Shortly thereafter, both the proprietary and non-proprietary versions of the feedwater line rupture topical report (Reference 2) were transmitted to the NRC in Reference 6. An initial submittal for the NOTRUMP code (Reference 3) was provided to the NRC. As a result of the NRC review of WCAP-9230 and WCAP-9236, requests for additional information were obtained from the NRC (Reference 7) in May 1980. Responses to these questions were provided in Reference 8 in August 1980.

Even prior to the generic approval received on the LOFTRAN computer code (1983), the NRC accepted the use of LOFTRAN in support of individual Safety Analysis Report submittals, including analyses of feedwater line rupture events which applied the methodology of WCAP-9230. In the NRC approval of the LOFTRAN code, the feedwater line rupture event is identified as an event for which LOFTRAN is an acceptable code to use in performing SAR analyses.

With respect to the use and interaction of the LOFTRAN and NOTRUMP codes, the following summaries of References 1, 2, and 8 are provided.:

- NOTRUMP models only the steam generator but provides adequate nodalization (See Appendix B of Reference 1 for the Model D steam generator) and includes modeling of the reference leg for level measurement based on the differential pressure from the level measurement taps.
- Boundary conditions must be defined for the NOTRUMP model which include steam flow, feedwater flow or break area to be used, and the primary system inlet conditions (e.g., RCS flow, pressure, and enthalpy). LOFTRAN can be used to define the primary system conditions and steamline pressure downstream of the steam generator nozzle.
- NOTRUMP provides predictions of the break flow conditions. A "predicted" or "bounding" quality of the break flow can be input to LOFTRAN in order to simulate a best-estimate or conservative LOFTRAN transient for the feedwater line rupture analysis. For the BVPS feedwater line rupture analyses, a bounding break quality is assumed.

As is noted in the response to Question 2, an issue has been entered into the Westinghouse Corrective Action Process (CAP) to assess the appropriateness of several assumptions made in the Westinghouse feedwater line rupture methodology for feedring type steam generators. The break discharge quality as a function of steam generator water level and interactions between LOFTRAN and NOTRUMP will be part of that investigation.

Figure 2-3 shows that acceptable analysis results are obtained for the BVPS units regardless of the assumed break discharge quality. The case represented in the plot is the limiting BVPS-2 case (0.717 ft2 break, with offsite power available) with the break quality from the time the feedring uncovers until dryout varied from zero (saturated steam) to one (saturated water).

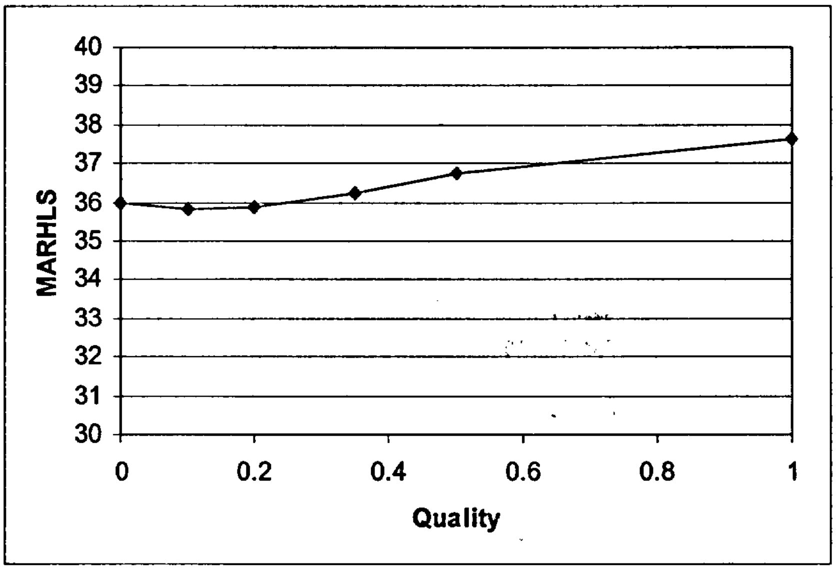


Figure 2-3

b. **UFSAR Section 15.2.8.2.1 provides a pressurizer safety valve (PSV) operability assessment for the FWLB event. In its July 8, 2005, response to RAI U.3.c, the licensee stated that this section of the UFSAR will be deleted because it no longer was applicable to BVPS-1 and 2.**

1. **Discuss the licensing basis and justification for deleting the UFSAR section identified above.**

Response

Section 15.2.8.2.1 was added to the BVPS-2 UFSAR in 1999. This section describes an analysis which was performed to evaluate the potential for liquid discharge during a Feedwater Line Break (FWLB) event. The analysis was performed following identification that evaluations performed to demonstrate compliance with NUREG 0737 Item II.D.1 based on the methodology of WCAP 11677 were in error. BVPS-2 compliance with NUREG Item II.D.1 was based on WCAP 11677. This WCAP lists PSV fluid conditions for a FWLB event and PSV qualification based on EPRI test results. While BVPS-2 was not specifically discussed in this WCAP, reference to this information was accepted by the NRC as a basis for showing compliance with NUREG 0737 Item II.D.1 based on the fact that the fluid conditions were expected to be similar to BVPS-1 and the specific BVPS-2 PSV model was part of the EPRI test program. The WCAP categorized plants in terms of those for which FWLB results in water relief and those for which no water relief would be predicted in the licensing basis FWLB described in the UFSAR. Analysis of FWLB at EPU conditions has demonstrated that no water relief occurs and therefore, BVPS-2 would now fall into this category. Therefore, it is appropriate to remove this section of the UFSAR since no PSV qualification for water relief during a FWLB is required.

Section 5.4.13.4 of the BVPS-2 UFSAR discusses qualifications of the PSVs. This sections states "The EPRI test conditions envelope the BVPS-2 plant-specific design conditions for both anticipated operational occurrences and accident conditions, and the piping and pipe supports are designed to withstand the resulting calculated loads in accordance with the applicable code requirements." This general statement remains accurate although FWLB will no longer be considered as an event for which water relief through the PSVs must be considered. Section 15.5.1 of the BVPS-2 UFSAR describes the inadvertent operation of the emergency core cooling system (ECCS). This section was also modified in 1999 to add consideration of water relief through the PSVs. This section will remain in the UFSAR and qualification of the PSVs for fluid conditions associated with this event under EPU conditions has been completed. Qualification of the associated discharge piping for fluid conditions associated with this event under EPU conditions will be completed prior to implementation as identified in commitments 1 and 2 of FENOC Letter L-05-112 Enclosure 3.

2. **From the BVPS-1 RSG/EPU transient response, it appears that the action of the power-operated relief valves (PORVs) (coupled with operators increasing AFW flow) terminates the RCS pressure rebound. Please evaluate the same case without PORV action to determine if RCS pressure and pressurizer liquid level remain within acceptable limits.**

Response

With respect to overpressuization, the feedwater line break event is bounded by the Loss of Load (LOL)/ Turbine Trip (TT) events, in which assumptions are made to conservatively calculate the RCS and MSS pressure transients. For a feedwater line break event, turbine trip occurs following reactor trip, whereas for the LOL/TT event, the turbine trip is the initiating fault. Thus, the primary to secondary power mismatch and resultant RCS and MSS heat up and pressurization transients are always more severe for the LOL/TT event. For this reason, it is not necessary to calculate the maximum RCS or MSS pressures for the feedwater line break event. Thus, the feedwater line break analysis biases inputs to conservatively demonstrate that no bulk boiling occurs in the hot leg.

Also, from the transient plots provided in Section 5.3.17 of the RSG and EPU LARs, it can be seen that the PORVs have adequate capacity to limit the pressure increase such that the pressure does not increase beyond the PORV setpoint. The PSVs have greater capacity than the PORVs. Thus, a reanalysis with the PORVs not modeled would result in an increased peak pressure but would not significantly exceed the PSV opening pressure. The pressure relieving capacity of the PSVs is not challenged by a FWLB event but it is challenged for a LOL/TT event which makes LOL/TT the most limiting non-LOCA primary pressure event.

With respect to pressurizer filling, analysis of FWLB at EPU conditions has demonstrated that no water relief occurs. Therefore, the pressurizer liquid level remains within acceptable limits. A FWLB is a relatively slow transient. In the cases analyzed in support of the BVPS RSG and EPU submittals, the peak pressurizer water level occurs no sooner than 20 minutes into the transient with significant margin to overfill. At 15 minutes, operator actions are assumed to increase AFW to the intact steam generators. Thus, the time where the peak pressurizer water volumes are predicted is well past the point in time where the operator is expected to be taking action to control the plant within acceptable bounds. One of those actions would be to isolate or limit safety injection based on indications of the pressurizer water level.

c. **If the low SG pressure safety injection (SI) signal is being credited for actuating the AFW pumps, please provide the AFW system response time to this signal (this may be different from response time of the AFW system to a low SG level signal).**

Response

For BVPS-1, AFW actuation is assumed to result from a compensated low steam pressure SI signal with a 60 second delay time. The system response following a low steam pressure SI signal is the same as following a low-low steam generator water level signal. The low steam pressure SI signal is also credited for Steamline Isolation with an eight second delay and for Safety Injection with a 27 second delay (with power case) or 37 second delay (without power case). For BVPS-2, AFW actuation is from the low-low steam generator water level signal with a 60 second delay. The low steam pressure SI signal is credited for Steamline Isolation with an eight second delay. Safety Injection in the BVPS-2 analysis results from a low pressurizer pressure signal with 27 and 37 second delays for the with and without power cases, respectively.

d. **In reviewing the AFW flow characteristics of Figure U.7-1, from the July 8, 2005, RAI response, the NRC staff requests that you provide the AFW flow density as a function of time. The NRC staff notes that the AFW flow ramps up prior to crediting operator action at 924 seconds. Please explain the reason for the flow ramp upward?**

Response

For both BVPS units, the AFW flow rate is 125 gpm/intact generator for 15 minutes and then 200 gpm/intact steam generator for the remainder of the transient. The enthalpy of the AFW is equal to the main feedwater enthalpy (436 Btu/lbm) until the AFW system purges the "hot" feedwater from the feedwater lines. At this point the AFW enthalpy is reduced to 90.77 Btu/lbm (120°F). A maximum purge volume of 168 ft^3 (BVPS-1) and 125.7 ft^3 (BVPS-2) are assumed.

The increase in AFW mass flow rate is due to the density change associated with the completion of the purge of the "hot" AFW from the feedwater lines. Note that the model actually models the transition from "hot" to "cold" as a ramp. The purge time is calculated by the code and the transition begins when 95% of the purge time has elapsed and continues until the purge time is 105% of the calculated purge time. This was done for code stability reasons so as not to introduce a step change into the analysis.

This can be seen in Figures U-1, U-3, U-5 and U-7 from the July 8, 2005 RAI responses in FENOC Letter L-05-112. AFW starts at 125 gpm and 436 Btu/lbm one minute after the actuation signal is generated. Once the "hot" AFW is purged, the AFW mass flow rate increases from about 14 lbm/sec to about 17 lbm/sec as the AFW enthalpy decreases from 436 Btu/lbm to 90.77 Btu/lbm as described in the previous paragraph. The final AFW mass flow increase occurs 15 minutes after the actuation signal via operator action as the volumetric flow increases from 125 gpm/intact generator to 200 gpm/intact generator.

3. Question

The NRC staff's SE dated December 1, 1998, states that a utility's use of the WRB-2M correlation with a departure from nucleate boiling (DNB) ratio limit of 1.14 for plant safety analyses, as described in approved topical report, WCAP-15025-P-A, may be approved and used provided the specified four conditions are met. Condition 4 states the correlation should not be used outside its range of applicability defined by the range of the test data from which it was developed. The NRC staff requests that the licensee provide a table with the data that demonstrates that the use of this correlation at BVPS-1 is within the range of parameters stated in WCAP-15025-P-A, Table 4-1, and that condition 4 of the December 1, 1998, SE for use of the WRB-2M correlation has been met.

Response:

In using the WRB-2M correlation for the RSG and EPU LARs, the analyses were done for several postulated accidents (e.g. Loss of Flow, Locked Rotor, and etc.). In performing the analysis, numerous statepoints for each postulated accidents were evaluated with the WRB-2M correlation. The parameters were confirmed to be within the applicable range of parameters stated in WCAP-15025-P-A Table 4-1.

WCAP-15025-P-A Table 4-1 is given below.

Parameter	Range
Pressure (psia)	$1495 \le P \le 2425$
Mass Velocity (Mlbm/hr-ft^2)	$0.97 \le G \le 3.1$
Quality	$-0.1 \le X \le 0.29$
Heat Length, inlet to CHF location (ft)	$Lh \le 14$
Grid Spacing (in)	$10 \le gsp \le 20.6$
Equivalent hydraulic diameter (in)	$0.37 \le De \le 0.46$
Equivalent heated diameter (in)	$0.46 \le Dh \le 0.54$

The following table presents the geometry parameters of the RFA/RFA2 fuel.

Heat length (ft)	≤ 12
Grid Spacing (in)	$10 \le gsp \le 20.6$
Equivalent hydraulic diameter (in)	0.438
Equivalent heated diameter (in)	0.491

The following table provides the parameters under some representative conditions where the minimum DNBR values were calculated.

Conditions	Pressure (psia)	Mass Velocity (Mlbm/hr-ft^2)	Quality
Complete Loss Of Flow (LOF)	2260	1.284	0.116
LOF Frequency Decay	2260	1.122	0.174
Partial LOF	2260	1.827	-0.001
Locked Rotor	2260	0.985	0.198
Core limits at Low Pressure and 100% Power	1940	1.629	0.262
Core limits at High Pressure and 100% Power	2425	1.615	0.247

4. Question

Technical specification change no. 8 listed in the licensee's April 13, 2005, RSG LAR consists of modifying the overpower delta temperature (OPΔT) and the overtemperature delta temperature (OTΔT) equations in TS 3.3.1.1, "Reactor Trip System Instrumentation." The NRC staff requests that the licensee provide the technical justification for why it is acceptable to add the lag compensators, yet leave out the lead/lag compensation function on both sides of the equation. Was the omission of the function and physical installation on the original design of the plant compensated for originally, and is there an effect on the plant that should be accounted for as an effect of this amendment?

Response:

The original design for BVPS-1 did not include all of the lag or lead/lag compensators of the standard equations used in later Westinghouse designs. The original design meets the requirements of the safety analyses, and no compensation was required for the lag or lead/lag compensators not originally installed. The safety analyses supporting the RSG and EPU LARs incorporates the use of lag compensators that were standard hardware design for latter Westinghouse plants. BVPS-1 has therefore, chosen to incorporate the lag compensator hardware necessary to reflect the safety analyses requirements.

The safety analyses for the RSG and EPU submittals do not require the lead/lag compensators left out on both sides of the equation, as the analyses assume the values are set to zero. The reason for BVPS-1 not having these lead/lag functions was that the original safety analyses did not credit or provide for any compensation for these lead/lag functions, which at the original licensing were not part of the design and were not required.

Without the lead/lag compensators BVPS-1 will be in compliance with the safety analyses for RSG and EPU. There are no BVPS-1 required hardware modifications for these compensators as the signals pass directly to the next module. BVPS-2, which has the later standard Westinghouse design, must have the lead/lag functions recalibrated to zero, effectively providing a pass through of the signal, producing the same circuit response as BVPS-1. Both BVPS-1 and BVPS-2 will be in compliance with the safety analyses for the RSG and EPU submittals with the addition of the lag compensators for BVPS-1 and rescaling of BVPS-1 and BVPS-2 instrumentation process racks. There are no additional effects on either plant that should be accounted for due to these amendments beyond those identified here.

The overall effect is that BVPS-1 and BVPS-2 are mathematically equivalent equations and both equations reflect the requirements of the safety analyses.

The proposed Overtemperature ΔT and Overpower ΔT coefficient changes reflect the analyses documented in Enclosure 2, Sections 5.3 and 5.10 of the RSG and EPU LARs, and are shown in Table 5.10-1. Incorporation of these changes supports operation at the EPU conditions for both BVPS-1 and BVPS-2. BVPS-1 operation with RSG at the existing power level is also supported by incorporation of these changes.

5. Question

Section 5.4 of the April 13, 2005, RSG LAR states that an operational response analysis of steam generator tube rupture (SGTR) was performed for BVPS-1. The NRC staff requests that the licensee provide a table listing the sequence of events and times from break initiation to event termination that shows operators can terminate the break flow from the ruptured SG within 51 minutes of accident initiation for the SG replacement and extended power uprate (EPU) conditions. Additionally, the staff requests that the licensee provide the results of the SGTR thermal-hydraulic analysis over time showing the pressurizer pressure, intact and ruptured SGs pressures, and ruptured SG water volume for the analysis to demonstrate that no overfilling of the SG occurs.

Response:

The SGTR operational response analysis for EPU conditions with RSG included cases to provide thermal-hydraulic tube rupture data for use in radiological dose consequence analysis and for evaluation of margin to overfill. The sequence of events for these cases are shown in Tables 5-1 and 5-2, respectively. The sequence of events tables show that termination of the event (i.e., termination of primary-to-secondary break flow) occurs at 3652 seconds (approximately 61 minutes) and 3578 seconds (approximately 60 minutes), respectively, for EPU conditions with RSGs. Figures 5-1 through 5-6 show the response of pressurizer level, pressurizer pressure, intact and ruptured SG pressure, intact loop RCS temperatures, primary-to-secondary break flow, and ruptured SG water volume for the margin to overfill case. The response to Question 6 provides information regarding the analysis performed in the 1990's for current power conditions that established a break flow termination time of 51 minutes.

Table 5-1
BVPS-1 Thermal-Hydraulic Analysis Sequence of Events

Event (Thermal & Hydraulics for Doses)	Time (seconds)
Steam generator tube rupture	0
Reactor trip – Overtemperature Delta-T	120
Safety injection initiated	148
Isolate auxiliary feedwater to ruptured steam generator	573
Ruptured steam generator steamline isolated	1122
Ruptured steam generator atmospheric dump valve (ADV) fails open	1124
Ruptured steam generator ADV block valve closed	1514
Reactor coolant system cooldown initiated	1660
Reactor coolant system cooldown terminated	2624
Reactor coolant system depressurization initiated	2804
Reactor coolant system depressurization terminated	2900
Safety injection terminated	3194
Steam relief to maintain sub-cooling	3264
Break flow terminated	3652

Table 5-2
BVPS-1 Margin to Overfill Analysis Sequence of Events

Event (Margin to Overfill)	Time (seconds)
Steam generator tube rupture	0
Reactor trip – Overtemperature Delta-T	106
Auxiliary feedwater initiated	121
Safety injection initiated	197
Isolate auxiliary feedwater to ruptured steam generator	515
Ruptured steam generator steamline isolated	1110
Reactor coolant system cooldown initiated	1254
Reactor coolant system cooldown terminated	2466
Reactor coolant system depressurization initiated	2646
Reactor coolant system depressurization terminated	2752
Steam relief to maintain subcooling	2952
Safety injection terminated	3046
Break flow terminated	3578
	Volume (ft^3)
Available ruptured steam generator secondary volume	5630
Maximum ruptured steam generator secondary volume	5602
Available ruptured steam generator secondary volume to overfill	28

Figure 5-1
Pressurizer Level

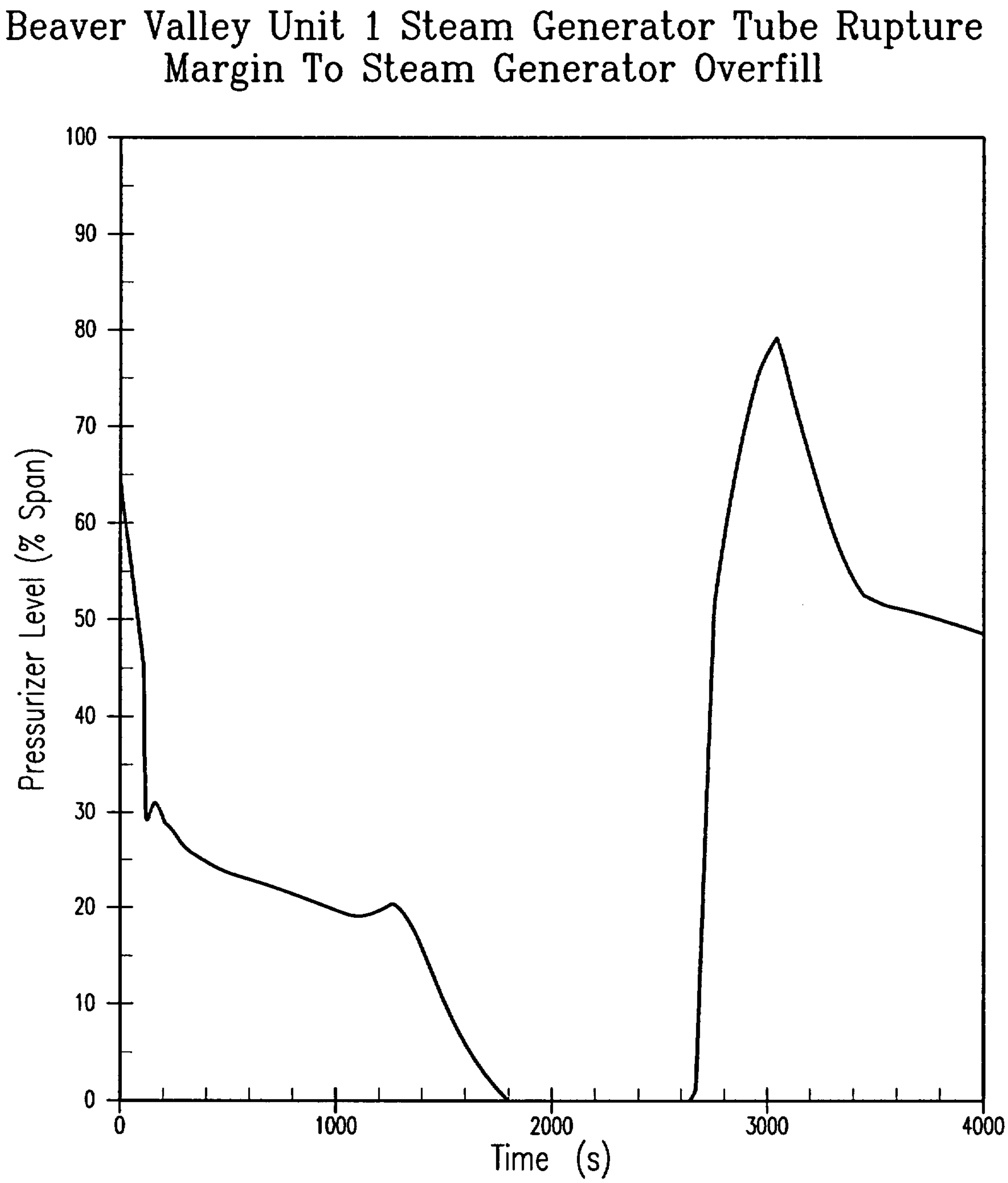


Figure 5-2
Pressurizer Pressure

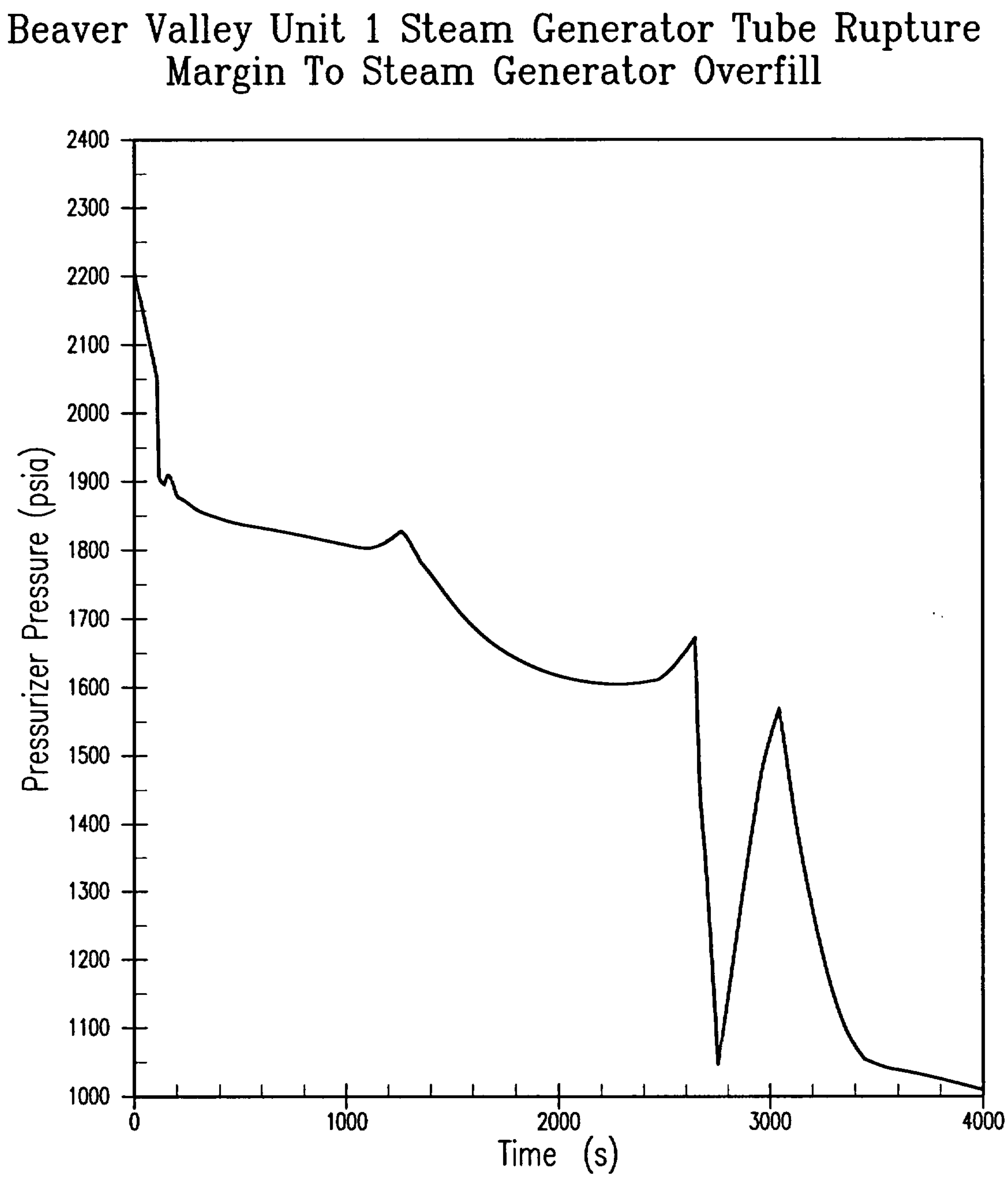


Figure 5-3
Secondary Pressure

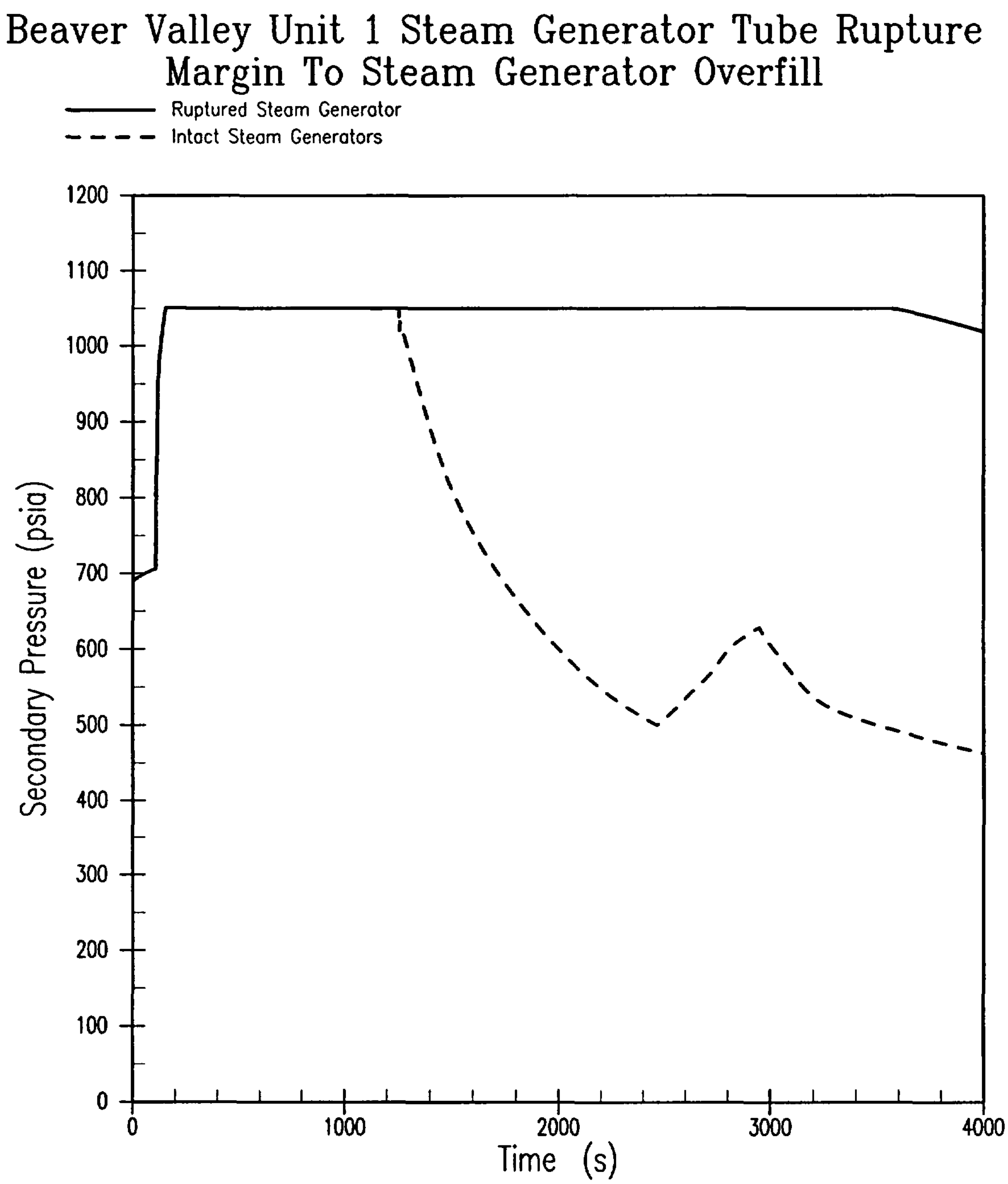


Figure 5-4
Intact Loop RCS Temperatures

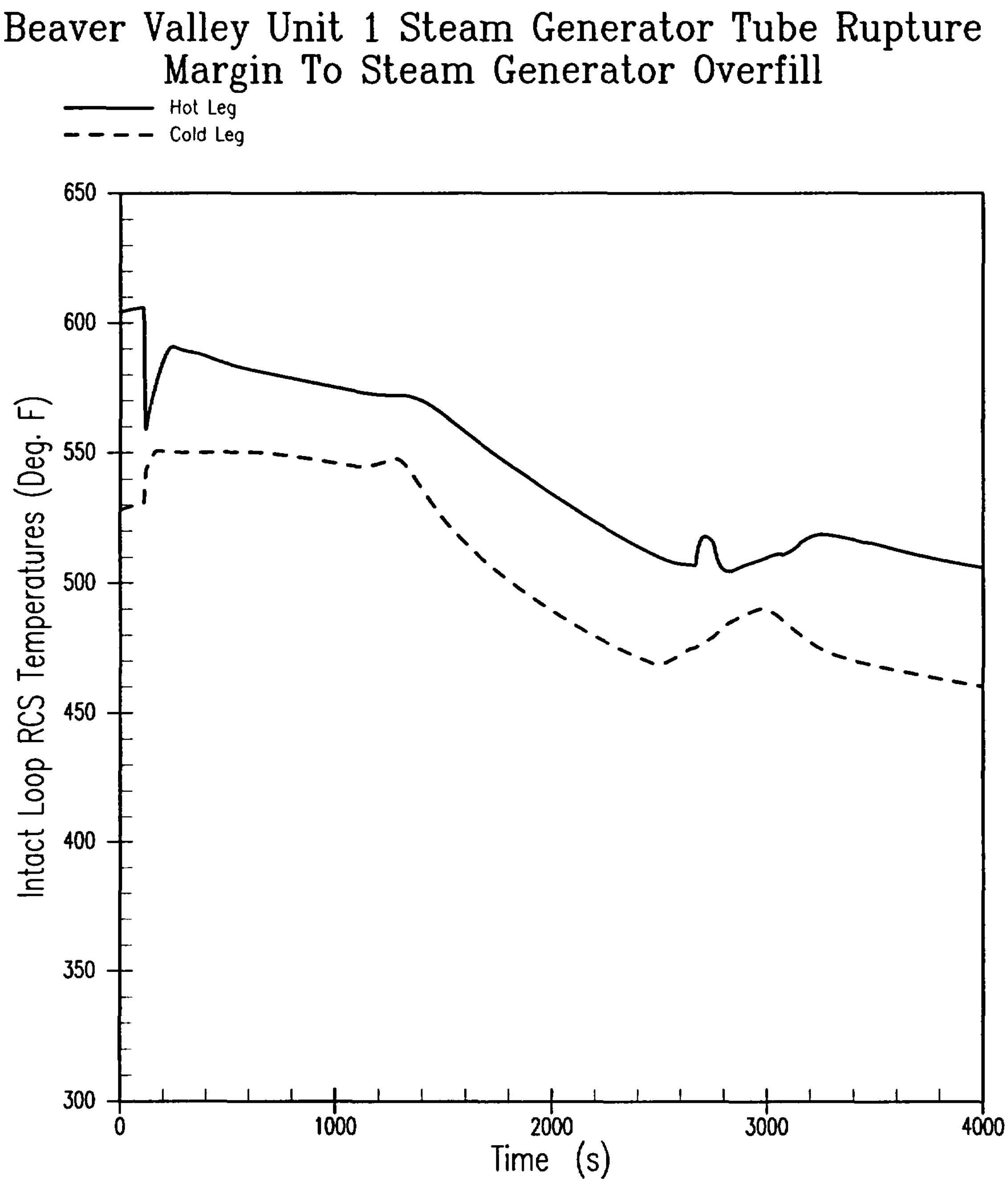


Figure 5-5
Primary-to-Secondary Break Flow

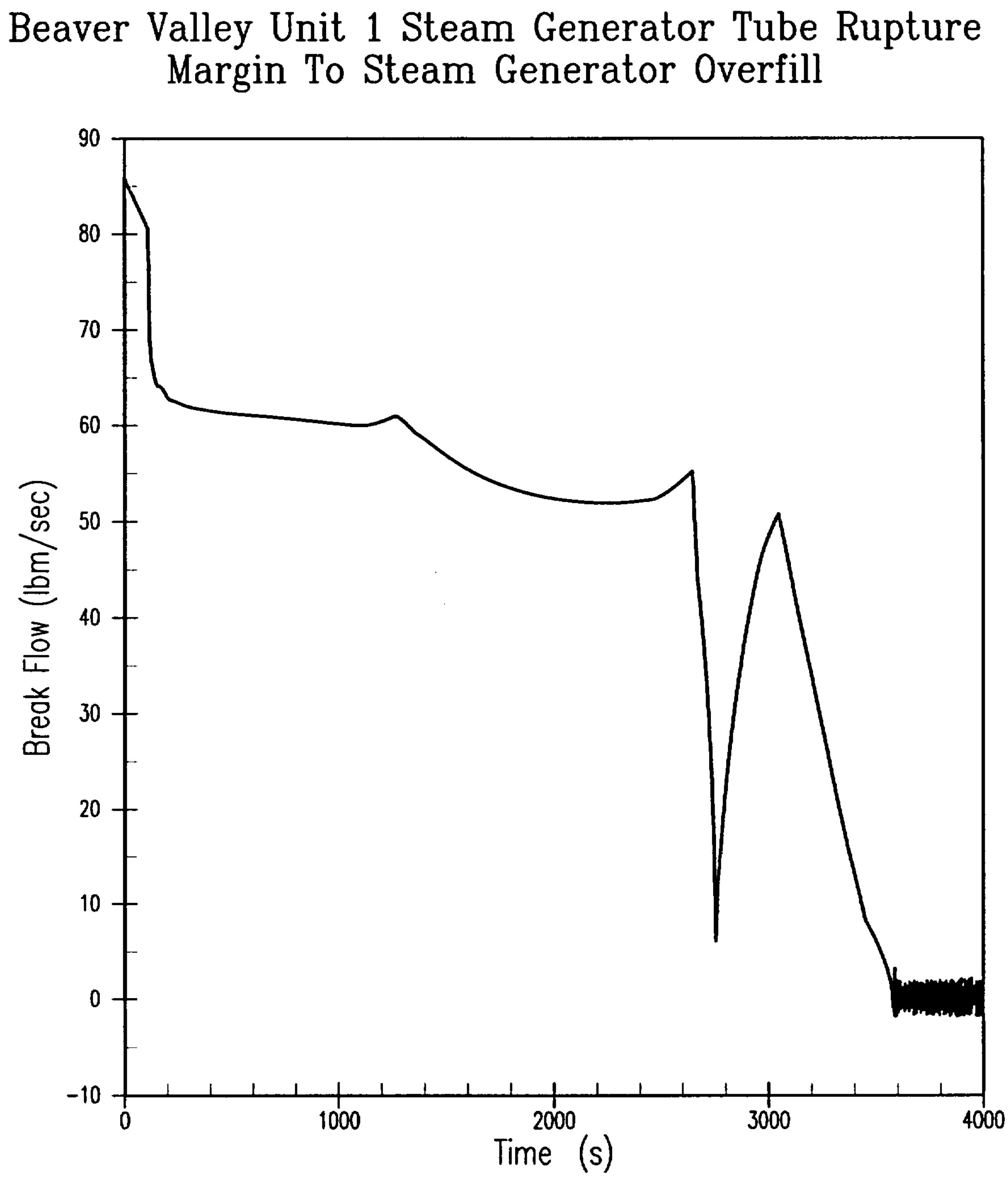
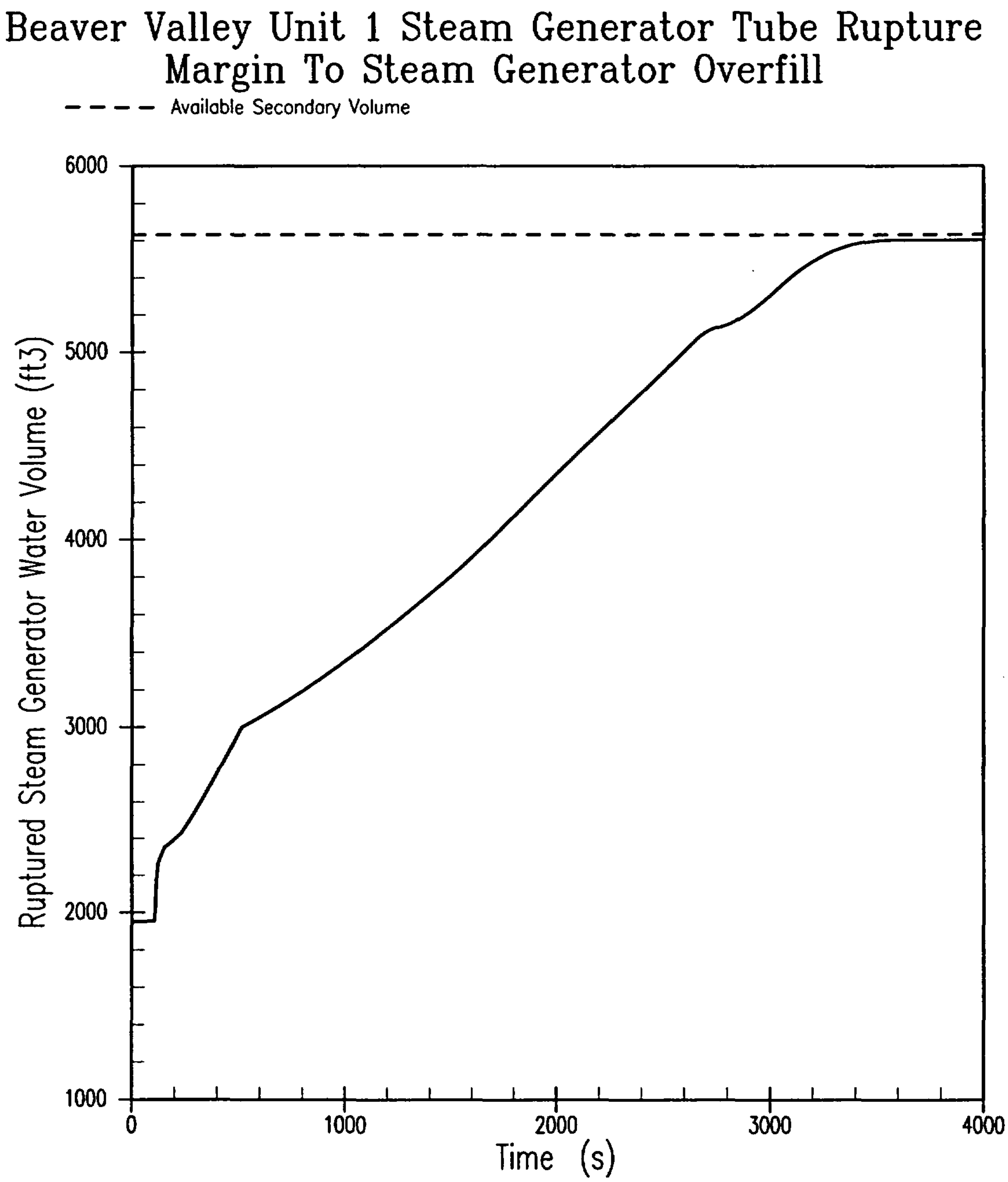


Figure 5-6
Ruptured Steam Generator Water Volume



6. Question

The NRC staff requests that the licensee provide the technical justification as to why a transient break flow analysis was not performed for the SGTR and why the 30-minute release assumption is conservative for the analysis when termination of the event exceeds 30 minutes.

Response:

The BVPS-1 SGTR licensing basis analysis methodology (Section 5.4 of Enclosure 2 of FENOC Letter L-05-069) consists of a thermal-hydraulic analysis to provide tube rupture data (e.g., break flow and steam releases) as input to the BVPS-1 SGTR radiological dose consequence analysis. This licensing basis methodology includes an assumption that the break flow and steam release from the ruptured steam generator are terminated at 30 minutes.

As described in the response to NRC RAI X.1 of Enclosure 2 of the July 8, 2005 (L-05-112) Response to RAIs on the EPU LAR, a condition report was written in the 1990's that documented that more than 30 minutes was required to terminate radioactive steam release from the ruptured steam generator. At that time through the corrective action process, the break flow termination time was revised to 51 minutes. Even though the break flow termination time increased, it was determined that the primary-to-secondary break flow based on the assumptions that terminate break flow in 30 minutes actually resulted in a higher primary-to-secondary break flow than the case that terminated break flow in 51 minutes. Consequently, it was concluded that the licensing basis methodology including the assumption of break flow termination at 30 minutes was conservative with respect to a transient break flow analysis that included a longer break flow termination time.

This approach established in the 1990's was retained for the BVPS-1 SGTR analysis for EPU conditions with the RSGs. A SGTR licensing basis methodology analysis was performed including the 30-minute isolation time to provide conservative break flow and steam release data as input to the SGTR radiological dose consequence analysis. The results of this SGTR licensing basis methodology analysis are presented in Section 5.4.1 of Enclosure 2 of the RSG LAR.

To develop operator action information for operator training as well as to confirm that the licensing basis methodology analysis continues to provide conservative estimates for the radiological dose consequences following a SGTR, a SGTR operational response (transient break flow) analysis was performed for BVPS-1 at EPU conditions with the RSGs. As noted in Section 5.4 of Enclosure 2 of the April 13, 2005 (L-05-069) RSG LAR, this operational response analysis demonstrates that the BVPS-1 SGTR licensing basis analysis methodology is conservative.

The SGTR radiological dose consequence analysis using the tube rupture data (e.g., break flow and steam releases) from the BVPS-1 SGTR operational response (transient break flow) analysis was provided in response to NRC RAI B.1 of Attachment A of the August 26, 2005 (L-05-137) Response to RAIs on the RSG LAR. This SGTR radiological dose consequence analysis demonstrates that the SGTR licensing basis methodology analysis with a break flow termination time of 30 minutes is more limiting than the operational response analysis with a break flow termination time of 3652 seconds (approximately 61 minutes), which is the break flow termination time calculated for the SGTR operational response analysis radiological dose consequence case at EPU conditions with RSGs.

The margin to overfill analysis from the operational response (transient break flow) analysis is provided in the response to Question 5. This margin to overfill analysis demonstrates that the primary-to-secondary break flow into the ruptured steam generator is terminated prior to overfilling the ruptured steam generator. The operational response analysis for the margin to overfill case at EPU conditions with RSGs shows a break flow termination time of 3578 seconds (approximately 60 minutes). The operator action times associated with this analysis have been provided as input to operator training. Additional information pertaining to operator actions and operator training has been provided in Enclosure 4 of the October 7, 2005 (L-05-154) EPU LAR Supplemental Information.

This information along with the more detailed information in the referenced RAIs provide the technical justification as to why the 30-minute SGTR licensing basis methodology analysis is retained for the BVPS-1 SGTR and why the 30-minute release assumption is conservative for the radiological dose consequence analysis when termination of the event exceeds 30 minutes. The technical justification includes a SGTR operational response (transient break flow) analysis as described in this response.

Note that the operator response times for the BVPS-1 SGTR operational response analysis are in the process of being validated as part of the emergency operating procedure (EOP) simulator validation process. One operator action time included in the SGTR operational response analysis case to provide thermal-hydraulic tube rupture data for use in radiological dose consequence analysis is the "local isolation of a failed-open atmospheric dump valve on the ruptured SG within 6.5 minutes after the valve fails open." The atmospheric dump valve (ADV) is assumed to fail open when the main steam line isolation valve for the ruptured SG is closed. This operator action time is being validated and is being tracked in the BVPS Corrective Action Program. In order to afford additional operator response time for this local operator action, additional analysis is being performed to allow up to a 10-minute operator action time. The preliminary analysis results for 10-minute operator action time still support the conclusion that the BVPS-1 SGTR licensing basis methodology with a break flow termination time of 30 minutes is more limiting with respect to radiological dose consequences than the operational response analysis. The results of the revised SGTR radiological dose consequence analysis and the validation of the 10-minute operator response time are scheduled for review at the November 29, 2005 NRC audit of the EPU/RSG radiological dose consequence analyses. This change to the operator action time to locally isolate a failed open ADV on the ruptured SG for the SGTR operational response analysis dose case does not impact the SGTR operational response analysis margin to overfill case.

7. Question

Table 5.3.20-1A states that a generic (W) evaluation addresses peak pressures for the rod withdrawal-at-power accident analysis. In the licensee's response to RAI H.3 of Section 5.3.3 of L-05-112, the licensee references this generic W report again. Please state the name of the W report being referenced, and confirm that it is applicable to BVPS-1.

Response:

The generic analysis referred to in FENOC Letter L-05-112, Enclosure 1 is an internal Westinghouse analysis. The Rod Withdrawal at Power (RWAP) analysis is performed for the primary purpose of demonstrating the adequacy of the High Neutron Flux and OTΔT reactor trip functions in preventing the violation of the DNBR safety analysis limit. For that reason, pressurizer pressure control mechanisms, such as pressurizer sprays and power-operated relief valves (PORVs), are assumed to operate as designed to minimize the calculated DNBR for the event. To address RCS overpressurization concerns, Westinghouse performed generic analyses without assuming operation of these control systems which utilized bounding values for several key input parameters. The generic analyses were performed using the LOFTRAN code which has been approved by the NRC for analyzing rod withdrawal and primary system overpressurization events. The results of this generic analysis, which considers 2-loop, 3-loop and 4-loop Westinghouse-designed plants, demonstrate that adequate protection would be provided through the use of the high neutron flux and high pressurizer pressure reactor trip functions in conjunction with the positive flux rate trip (PFRT) reactor trip function. This last function is typically not explicitly modeled in safety analysis since most utilities do not perform response time testing on it.

However, the generic work performed to address RWAP overpressurization takes credit for the function's availability (confirmed by surveillance requirements that ensure its operability) without placing any requirement on its response time. The generic work concluded that the presence of the protection functions noted above ensures that overpressurization following a RWAP, assuming all automatic pressure control features are unavailable, is bounded by other Condition II transients. As such, the analysis methodology for this event focuses on the limiting acceptance criterion for this event (i.e., DBNR) and assumes both pressurizer sprays and PORVs are available.

A review of the key input assumptions made in the generic overpressurization analysis was performed as part of the BVPS-1 RSG and EPU submittal preparation, and it was confirmed that the generic analysis applies to BVPS-1 with the RSGs. Below is a table showing the key input assumptions that were compared in order to conclude that the generic analysis is applicable and bounding for BVPS-1.

	Generic Analysis Assumptions	BVPS-1 RSG
	3-Loop	3-loop
Core Power (MWt)	2910	2910
Power Uncertainty (%)	2 *	0.6
Nominal RCS Pressure (psia)	2250	2250
Pressure Uncertainty (psi)	± 40	± 40
Nominal Full Power Vessel Tavg (°F)	580.0	580.0
Temperature Uncertainty (°F)	+8.5	+8.5
High Flux Trip Setpoint (%)	118 *	116
High Flux Trip Delay (sec.)	0.5	0.5
High Pressurizer Pressure Trip Setpoint (psia)	2435	2435
High Pressurizer Pressure Trip Delay (sec.)	2.0	2.0
Positive Flux Rate Trip Setpoint (%/sec.)	9.0 *	5.96
Positive Flux Rate Trip Time Constant (sec.)	2.0	2.0
Positive Flux Rate Trip Delay (sec.)	3.0	N/A
PSV Setpoint (psia)	2500	2500
PSV Tolerance (%)	±4 *	±3
PSV Loop Seal Delay (sec)	1.5 *	0.33
Max. Reactivity Insertion Rate (pcm/sec.)	110.0	110.0

* More conservative than BVPS. Analysis with BVPS data rather than the generic data would yield less limiting results.

It should be noted that the generic method described above has been reviewed by the NRC in a submittal for the Diablo Canyon Nuclear Plant.

8. **Question**

In Section 5.6.2.2.13, "Core Decay Heat," of the April 13, 2005, RSG LAR, the licensee states that the core decay heat generation assumed in calculating the MSLB mass and energy (M&E) release is based on the 1979 American Nuclear Society's (ANS's) Decay Heat + 2σ model. The licensee states that this is a deviation from the current licensing-basis analysis of MSLB M&E releases outside containment for BVPS-1 which uses the 1971 standard (+20% uncertainty). For the NRC staff to review your use of the 1979 ANS Decay Heat model, please provide the following additional information:

a. Tabulated Decay Heat Curve used in the analysis

Response:

a,c

b. **Assumptions used in tabulating your decay heat curve, and a basis for each assumption. These should include:**

(1) **U-238 Fission Fraction**
(2) **U-235 Fission Fraction**
(3) **Pu-239 Fission Fraction**
(4) **Other heavy element decay heat sources and their contributions to the total decay heat power**
(5) **Total recoverable energy associated with one fission and the standard deviation of this value**
(6) **Length of time fuel is assumed to be at full power**
(7) **Inputs into the calculation of the factor which accounts for neutron capture, or state if Table 10 of the 1979 ANS standard was used**

Response

These fission fractions are based on actual U-238, U-235, and Pu-239 fissions that would occur for fresh fuel load enrichment levels varying from 1.5% to 5% with the fission fractions dependent upon the burnup of the fuel.

U-239 and Np-239 have also been included in the calculation of the decay heat power. The contributions are based on Equations 14 and 15 from ANSI/ANS-5.1-1979.

The total recoverable energy per fission is assumed to be 200 MeV for all isotopes. The standard deviation is 1.5% for all isotopes.

The decay heat values bound any finite irradiation time equivalent to constant full-power operation to reach a variety of burnup levels (effective region-average burnup levels) in the fuel. Burnup levels included in the model range from 1000 MWD/MTU to 60,000 MWD/MTU. The upper limit on burnup varies dependent upon the assumed initial enrichment of the fuel as noted above.

Neutron capture is defined by Equation 11 from ANSI/ANS-5.1-1979 for shutdown times of less than 10,000 seconds and Table 10 from ANSI/ANS-5.1-1979 for shutdown times of greater than 10,000 seconds.

The following questions refer to the licensee's RAI response dated July 8, 2005, related to the EPU LAR.

Steam System Piping Failure at Full Power

9. Question

RAI W.7 requested the licensee to demonstrate that inclusion of a single failure and/or loss of offsite power (LOOP) would not create a more severe scenario. In response to RAI W.7, the licensee stated that a LOOP was considered in the post-trip hot zero power (HZP) MSLB event but was less limiting than the case with full RCS flow.

a. **The timing of a LOOP is different for the locked-rotor event and the feedwater line break event in the BVPS-1 and 2 UFSARs. Define the licensing basis for the timing of the LOOP relative to the reactor trip signal.**

Response

Consistent with the Westinghouse generic guidelines for the feedwater line break analysis, a 2-second delay for the loss of the reactor coolant pumps (RCPs) is assumed for the BVPS feedwater line break analyses (both units, current power and EPU). This is considered a reasonable delay to use to demonstrate the differences between the cases analyzing the long term consequences of the feedwater line break event for variations in the availability of offsite power. A one, two or ten second delay would have no significant impact on the results since the analysis is demonstrating the heat removal capabilities of the AFW with the RCS in natural circulation conditions.

Consistent with the Westinghouse generic guidelines for the locked rotor safety analysis, the BVPS locked rotor analyses (both units, current power and EPU) conservatively assume no delay for the LOOP (and no delay for the turbine trip, generator trip, or automatic bus transfer to offsite power of the RCPs). Thus, reactor trip, turbine trip and RCP trip occur at the same time. This dramatically decreases the heat removal capacity of the plant and yields much more severe analysis results.

b. **Demonstrate that a LOOP would not create a more limiting scenario for the hot full power (HFP) MSLB event.**

Response

For the HFP main steamline break, the only significant impact of the LOOP would be the eventual loss of power to the RCPs causing a flow coast down to occur. Consistent with the Westinghouse generic requirements for the reactor protection system design, FENOC has implemented a (minimum) 30 second delay following an RPS initiated turbine trip before automatic bus transfer to offsite power is attempted for the RCPs at BVPS. This delays the potential for RCP coast down until well after the HFP MSLB event is terminated. For example, in the limiting case for BVPS-1, peak heat flux and minimum DNBR occur within a half a second of the reactor trip. Initiating a LOOP 30 seconds after trip would have no affect on the minimum DNBR or peak heat flux.

10. Question

In the EPU LAR, DNB is used as the sole criteria for clad failure. With the rapid power excursion associated with the pre-trip MSLB event (beyond 120%), the licensee must demonstrate that clad failure does not occur as a result of excessive clad strain due to fuel thermal expansion or incipient fuel melt. Demonstrate that local power peaking does not produce clad failure as a result of clad strain or quantify the amount of clad failure and provide an estimate of the radiological consequences.

Response

The design criteria that are currently used for the Hot Full Power Main Steamline Break (HFP MSLB) event correspond to the ANS Condition II criteria. Although a MSLB is classified as a Condition III/IV event, depending on the size of the break, the Condition II criteria are conservative with respect to the Condition III/IV event criteria. The criteria for fuel rod design are that the fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate (expressed in kW/ft) should not exceed a value which would cause fuel centerline melting. Based on the thermal hydraulic design calculations for the EPU LAR, the peak linear heat generation rate limit is 22.4 kW/ft. This limit has been shown to be met for the HFP MSLB event. Calculations have been completed for the EPU to confirm that the clad stress criterion has been met. Clad strain is evaluated by using the same methods as for clad stress, and has been shown to be less limiting than clad stress. Therefore the clad strain criterion has been shown to be met.

Steam System Piping Failure at HZP - Post-Trip Return-to-Power

11. Question

RAI Q.16 requested the licensee to demonstrate that clad failure does not occur as a result of high local power density. In response, the licensee stated that DNB has been taken to be the necessary and sufficient demonstration that fuel clad failure does not occur. The NRC staff does not accept this position. With the high local power peaking factors beneath the stuck-out control rod and a predicted post-trip return-to-power, the licensee must demonstrate that clad failure does not occur as a result of excessive clad strain due to fuel thermal expansion or incipient fuel melt. Demonstrate that local power peaking does not produce clad failure as a result of clad strain or quantify the amount of clad failure and provide an estimate of the radiological consequences.

Response

The criteria for fuel rod design for the Hot Zero Power Main Steamline Break (HZP MSLB) are that the fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate (expressed in kW/ft) should not exceed a value which would cause fuel centerline melting. Based on the thermal hydraulic design calculations for the EPU LAR, the peak linear heat generation rate limit is 22.4 kW/ft. Calculations demonstrate that this limit has been met for the HZP MSLB event. Calculations have been completed for the EPU to confirm that the clad stress criterion has been met. Clad strain is evaluated by using the same methods as for clad stress, and has been shown to be less limiting than clad stress. Therefore, the clad strain criterion has been shown to be met.

12. Question

In response to RAI Q.17, the licensee stated that a DNB ratio (DNBR) multiplier of 0.88 was used with the W-3 correlation. Please provide further discussion on the basis of the 0.88 multiplier.

Response

The basis of the 0.88 multiplier for the W-3 correlation to be used for 17x17 fuel assemblies is documented in WCAP-8296-P-A. The 17x17 fuel assembly design introduced changes in grid design and fuel rod geometry. Testing data showed that the Critical Heat Flux of 17x17 fuel assembly was over predicted by the W-3 correlation. A multiplier of 0.88 was thus determined and justified in WCAP-8296, providing good agreement with the 17x17 data. The NRC has reviewed and approved WCAP-8296-P-A (February 1975).

Boron Dilution

13. Question

In the response to RAI J.2 the licensee states, "The Mode 3 initial and critical concentrations are typically the most difficult to meet. Thus, in order to facilitate future reload evaluations, the Mode 3 initial and critical concentrations were adjusted so that the analytical limit of 15 minutes is met. Using overly conservative boron concentrations in the Mode 3 boron dilution calculation gives the core designers more flexibility during subsequent reload evaluations." Please provide a description of the adjustments that are made to the Mode 3 initial and critical concentrations and how they result in conservative boron concentrations in Mode 3.

Response

The previous response was not clearly stated. The intent of the previous response was to explain why the Mode 3 calculation appeared to be more limiting than the Modes 1 and 2 calculations. The following discussion explains how and why the Mode 3 initial and critical concentrations were adjusted and also how the final values are conservative.

Mode 3 boron dilution calculations for plants of the same vintage as BVPS are a hand calculation using the following relationship:

Dilution Time = (RCS Mass / Dilution Mass Flow Rate) * ln (critical conc./initial conc.)

The RCS mass and dilution mass flow rates are not cycle specific values. Only the initial and critical concentrations can vary from cycle to cycle due to the core design/loading pattern, etc. Also, in terms of demonstrating that the acceptance criterion is met, the important factor is the ratio of the critical concentration to the initial concentration. That is, critical and initial concentrations of 1500 ppm and 2000 ppm, respectively, will result in the same dilution time as 1575 ppm and 2100 ppm because they yield the same ratio.

In the past, Mode 3 calculations were performed using initial and critical concentrations determined by neutronics calculations. These values were conservative but not excessively conservative and resulted in dilution times well in excess of the acceptance criteria of 15 minutes. Thus, the margin was unavailable to address small variations in the core design from cycle to cycle. For the RSG and EPU analyses, it was decided to use initial and critical concentrations that resulted in less margin to the acceptance criteria of 15 minutes so the calculation was performed backwards to see what concentrations would yield acceptable results. This results in a UFSAR analysis that appears to be very limiting but actually has significant margin in the boron concentration inputs compared to the core design. This will eliminate the need to make small UFSAR changes for cycle specific variations.

14. Question

In the response to RAI A.1 the licensee provided Table A.1-9, "Uncontrolled Boron Dilution" (EPU Licensing Report Section 5.3.5), and in response to RAI A.5 the licensee provided Table A.5-1, "Licensing Basis Safety Analyses." In these tables several input parameters are described as nominal values. Explain how the use of these nominal values meets the requirement to use conservative values as indicated by the NRC review standard, RS-001, and NUREG-0800, "Standard Review Plan," Section 15.4.6.

Response

For Boron Dilution analyses, the dilution flow rate, initial boron concentration and critical boron concentration are conservative values. The initial and critical boron concentrations are re-confirmed to be conservative on a cycle-by-cycle basis as discussed in the response to Question 13. The active RCS volume used in the analysis represents a conservative combination of nominal volumes. For example, no credit is taken for the volume in the pressurizer, no credit is taken for thermal expansion of the RCS (piping, reactor vessel, etc.) and, in Mode 3, no credit is taken for the volume in the reactor vessel upper head. Also, specific volumes and densities used in defining the RCS mass and dilution mass flow rate are calculated at the extreme ends of the temperature ranges (including applicable uncertainties) for a given mode of operation. The calculation also assumes 30% of the steam generator tubes are plugged.

Additional RAI questions received on October 13, 2005:

15. Question

In response to staff RAI 5.3.3 H.3.a of L-05-112, Enclosure 1, FENOC references a Westinghouse generic analysis for the RCCA bank withdrawal at power event overpressurization concerns. Additionally, FENOC references an amendment specifically approved for Diablo Canyon (ML041180285). In this referenced document, it states a generic Westinghouse evaluation determined that a rod withdrawal at power could result exceeding the 110 percent design limit if only the typically credited UFSAR Chapter 15 trip functions (high pressurizer pressure, OTΔT, and power range high neutron flux) were credited and assuming that the pressurizer pressure control system malfunctions. Therefore, the licensee for Diablo Canyon evaluated the effects on RCS pressure on its plant crediting the positive flux rate trip to ensure the safety limits continue to be met. Please state the specific Westinghouse report FENOC is referencing and provide the plant specific analysis that shows the Westinghouse generic analysis is applicable to BVPS and overpressurization will not occur assuming the PORVs are not available.

Response

This question is very similar to Question 7. The generic analysis referred to in FENOC Letter L-05-112, Enclosure 1 is an internal Westinghouse analysis. The generic internal analysis that was performed assumed that the pressurizer pressure control systems (i.e., PORVs and sprays) did not operate and it uses bounding values for several key input parameters. The generic analysis considers 2-loop, 3-loop and 4-loop Westinghouse-designed plants. A comparison of key parameters in the generic 3-loop analysis to BVPS is tabulated in the response to Question 7.. The table clearly shows that the generic analysis is conservative compared to the BVPS units. The results of this generic analysis demonstrates that adequate protection would be provided through the use of the reactor trips from the neutron flux and pressurizer pressure signals.

The analysis concluded that the presence of the protection functions and safety valves ensures that overpressurization following a RWAP, assuming all automatic pressure control features are unavailable, is bounded by other Condition II transients. As such, the analysis methodology for this event focuses on the limiting acceptance criterion for this event (i.e., DBNR) and assumes both pressurizer sprays and PORVs are available.

16. Question

In response to staff RAI 5.3.5 J.2 of L-05-112 Enclosure 2, FENOC states the active RCS volumes assumed in Modes 1 and 2 include the upper head (7593 cubic feet) and the Mode 3 volume does not include the upper head (6964 cubic feet). Please provide an explanation of how the active RCS volume was determined for each mode and whether the replacement steam generators were considered in the calculation since the current values differ from the EPU values stated in Table A.1-9 of L-05-112 Enclosure 2.

Response

The BVPS-1 volumes include the Model 54F Replacement Steam Generators. For Modes 1 and 2, the active volume is calculated as follows:

Total Active Volume = Vrv + NLOOP(Vhl + Vsg + Vxo + Vpm +Vcl)

where: Vrv = reactor vessel volume including the upper head
NLOOP = the number of RCS loops
Vhl = hot leg piping volume
Vsg = steam generator primary side volume (accounting for plugged tubes)
Vxo = crossover leg piping volume
Vpm = reactor coolant pump volume
Vcl = cold leg piping volume

For Mode 3, the active volume is calculated exactly the same except that the volume in the upper head is deleted. In the Modes 1 and 2 calculations, with all reactor coolant pumps running, the flow rate to the upper head is approximately 500 gpm or 66.5 ft^3 per minute. With a total upper head volume of 629 ft^3, there is more than adequate flow for this volume to effectively contribute to the dilution volume seen by the core. A specific evaluation of this was performed to validate this inclusion of the upper head volume in the model for the Mode 1 and 2 analyses.

17. Question

In response to staff RAI 5.3.11 P.1 of L-05-112 Enclosure 2, FENOC states the analyses performed for BVPS-1 model only the low pressurizer pressure and OTΔT trip functions. Additionally, in section 5.3.11.5 of the RSG LAR, FENOC states the results of the analysis show that the pressurizer low pressure and OTΔT reactor protection system signals provide adequate protection against the RCS depressurization event since the minimum DNBR remains above the safety analysis limit throughout the transient. The staff requests FENOC provide the analysis that supports these statements and shows the OTΔT trip being credited performs its design function and provides adequate protection against DNB. The staff also requests FENOC provide the DNBR value obtained in the analysis that shows the safety analysis limit is not exceeded.

Response

Adequate protection for an RCS Depressurization event is provided via the low pressurizer pressure and OTΔT reactor trip functions (Section 5.3.11.5 of the RSG LAR). The OTΔT setpoints are validated for a window of conditions in temperature (Tavg), power (defined by delta-T) and RCS pressure space encompassing the nominal operating point (100% power, nominal Tavg, 2250 psia, nominal flow) and which ensure that the DNB design basis is satisfied. The RCS pressure range is limited by the low pressurizer pressure and high pressurizer pressure reactor trip setpoints, which ensure that a reactor trip will occur within the range of pressures that the DNB correlation is valid for, thereby ensuring that the OTΔT setpoints remain valid and provide protection against DNB. Thus, it is the combination of the low pressurizer pressure and the OTΔT reactor trip functions that provide protection for an RCS depressurization event and ensure that the DNB design basis is satisfied. Therefore, a reactor trip generated by either a low pressurizer pressure or OTΔT function will ensure that the DNB design basis is satisfied for the RCS depressurization event.

The calculated DNBRs are contained in the BVPS EPU Licensing Report in Tables 5.3.20-1A (BVPS-1) and 5.3.20-1B (BVPS-2). The calculated minimum DNBRs are 1.62 (BVPS-1) and 1.64 (BVPS-2). Both are compared to a limit value of 1.55.

References:

1. WCAP-9230 (Proprietary) and WCAP-9231 (Non-proprietary), "Report on the Consequences of a Postulated Main Feedline Rupture," January 1978.
2. WCAP-9236 (Proprietary) and WCAP-9237 (Non-proprietary), "NOTRUMP: A Nodal Transient Steam Generator and General Network Code," February 1978.
3. WCAP-8913, "NOTRUMP: A Nodal Transient Steam Generator and General Network Code," September 1977.
4. WCAP-7907 and WCAP-7907-P-A, "LOFTRAN Code Description," October 1972 and April 1984, respectively.
5. NRC Letter from J. F. Stolz, US NRC, to C. Eicheldinger, titled "Proposed Topical Reports," January 6, 1978.
6. Westinghouse letter, NS-CE-1671, T. M. Anderson (Westinghouse) to J. F. Stolz (US NRC), January 27, 1978.
7. NRC Letter from J. R. Miller to T. M. Anderson, titled, "Review of WCAP-9230," May 8, 1980.
8. Westinghouse letter, NS-TMA-2286, (Responses to NRC First Round Questions on WCAP-9230) T. M. Anderson to J. R. Miller, August 15, 1980.

L-05-165 Enclosure 2

Affidavit CAW-05-2067

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) 374-4419
Direct fax: (412) 374-4011
e-mail: maurerbf@westinghouse.com

Our ref: CAW-05-2067

November 16, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-MPG-05-133 P-Attachment, "Response to NRC Request for Additional Information on Replacement Steam Generators (RSGs) for Beaver Valley Power Station, Unit 1 (BVPS-1)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-05-2067 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 FirstEnergy Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2067, and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney
L. Feizollahi

CAW-05-2067

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer , 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:

BFMaurer

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 18th day
of November, 2005

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 Acting 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 in "brackets" in LTR-MPG-05-133 P-Attachment, "Response to NRC Request for Additional Information on Replacement Steam Generators (RSGs) for Beaver Valley Power Station, Unit 1 (BVPS-1)" (Proprietary) dated November 18, 2005, for Beaver Valley Power Station, being transmitted by the FirstEnergy Nuclear Operating Company FENOC letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Beaver Valley Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of Decay Heat Assumptions used for Main Steamline Break Mass and Energy Releases Outside Containment.

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

(a) Assist the customer in obtaining NRC approval by responding to NRC.

Further, this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for plant-specific applications for other customers.

(b) Its use by a competitor would improve his competitive position in the design and licensing of a similar product.

(c) The information requested to be withheld reveals specific aspects of safety analysis inputs which were 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 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).