FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
Route 168
P.O. Box 4
Shippingport, PA 15077-0004

Richard G. Mende
Director, Performance Improvement

724-682-5206

February 11, 2005
L-05-006

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

Subject: **Beaver Valley Power Station, Unit No. 1 and No. 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Response to a Request for Additional Information in Support of License Amendment Requests Nos. 317 and 190

This letter provides FirstEnergy Nuclear Operating Company (FENOC) responses to the December 14, 2004 NRC Request for Additional Information (RAI) regarding License Amendment Request (LAR) Nos. 317 and 190. The responses to the December 14, 2004 RAI are provided in Attachments A and B of this letter.

The subject LARs were submitted by FENOC letter L-04-073 dated June 2, 2004. The changes proposed by the LARs will revise the Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 Technical Specifications to permit each unit to be operated with an atmospheric containment design.

FENOC requests approval of the proposed amendments by June, 2005. However, since a number of the proposed Technical Specification changes require a plant outage to implement, FENOC requests the following implementation periods. The Unit 1 amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 1R17 refueling outage. The Unit 2 amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 2R12 refueling outage. Refueling outage 1R17 is planned for the spring of 2006 and refueling outage 2R12 is planned for the fall of 2006.

Attachment C lists the regulatory commitments made in this transmittal. If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Supervisor - Licensing, at 330-315-6944.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 11, 2005.

Sincerely,

Richard G. Mende

Attachments:

A Responses to December 14, 2004 Request for Additional Information
B MAAP-DBA One-Dimensional Heat Conduction in a Plane Wall
C Commitment List

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

L-05-006 Attachment A

REQUEST FOR ADDITIONAL INFORMATION
RELATED TO FIRSTENERGY NUCLEAR OPERATING COMPANY (FENOC)
BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)
CONTAINMENT CONVERSION TO ATMOSPHERIC CONDITIONS
DOCKET NOS. 50-334 AND 50-412

By letter dated June 2, 2004, FENOC (the licensee) proposed changes to Beaver Valley Power Station (BVPS-1 and 2) Technical Specifications (TSs) to allow operation of the containments at atmospheric conditions. The BVPS-1 and 2 containments are currently operated at subatmospheric conditions. In order for the staff to proceed with its review of the proposed change, the following information is needed. Underlined references refer to portions of the June 2, 2004, submittal and other references are included for ease of access to the information being sought.

1. Enclosure 2 Section 3.1.1 (a) What is the difference in methods and assumptions used in the Loss-of-Coolant-Accident (LOCA) containment peak pressure and temperature calculations and the LOCA containment response for the Net Positive Suction Head (NPSH) calculations, especially for BVPS-1 for which credit is taken for containment accident pressure in calculating available NPSH? (b) As part of the response to (a) please specify what conservative assumptions are made in calculating the containment accident pressure and sump water temperature as input to available NPSH calculations?

   Response:

   The difference in the methods and assumptions used in the LOCA containment peak pressure and temperature calculations and the LOCA containment response for the NPSH calculations was principally the use of the single node containment model (as discussed in Enclosure 2 Section 4.2.1) and the multiple node containment model (as discussed in Enclosure 2 Section 4.2.2).

   The design basis containment response calculations are implemented consistent with the intent of the Standard Review Plan. The containment peak pressure and temperature responses for large LOCA and main steam line breaks use the Tagami and Uchida heat transfer correlations to conservatively (understate) quantify the participation of the passive heat sinks. Because these correlations are derived from a single node representation of a vessel receiving the steam and water discharge from a blowdown, a consistent implementation of these heat transfer correlations leads to the use of a single node containment model. Thus, the total containment volume and passive containment heat sinks are incorporated in a single node containment model that is applied for quantifying the peak pressure, peak gas temperature, and maximum liner temperature for the spectrum of main steam line break and large LOCA breaks.

   Some of the assessments for the long-term containment response attributes are conducted with a multiple node containment model. Specifically, the large break LOCA NPSH, the small break LOCA NPSH, as well as the large and small break LOCA sump water temperature attributes use a multiple node model. The sump water level and temperature histories are key results to quantifying these specific attributes. Thus, the relative delivery rate and removal of water inventory from the containment sump and lower compartment influence the NPSH and sump temperature histories. Water hold-up from the break or spray injection sources in containment subcompartments directly influences the sump water level and temperature histories. Additionally, the distribution of containment sprays as they are collected on the operating deck floor can also influence these attributes. Thus, a multiple node containment configuration that identifies elevations and sizes of junctions connecting their various containment regions is implemented for these evaluations including the NPSH quantification. Similar to previous design basis evaluations, the BVPS-1 available NPSH method continues to include the overpressure credit, while the BVPS-2 available NPSH assessment does not credit overpressure. For these assessments, the multiple node model uses natural convection heat transfer models for calculating the energy transfer rate to the containment heat sinks distributed through these multiple nodes. Consistent with a design basis approach, the natural convection heat transfer models are biased to minimize the calculated available NPSH.

Table 4-3 of Enclosure 2 summarizes how the 24 initial conditions and parameter values were selected to conservatively bias the containment accident peak pressure and NPSH calculations, as well as the calculations for peak equipment qualification temperature, peak liner temperature, maximum sump water temperature and maximum service water outlet temperature (see Enclosure 2 Section 4.3). For example, the peak containment pressure assumes an initial containment pressure that is the maximum operating value whereas the available NPSH calculation assumes the minimum operating initial containment pressure. By this process the limiting parameters for the available NPSH calculation were applied. For example, Case 6L for BVPS-1 is a double-ended pump suction break with minimum safety injection. When this case was quantified for the peak pressure attribute (based on the parameter conditions identified in Enclosure 2, Table 4-3) the containment pressure peak was calculated to be 42.0 psig. However, when the available NPSH calculation was calculated with the parameters biased as indicated in Table 4-3 for this attribute Case 6L yielded a peak containment pressure of 40.7 psig. The lower overpressure condition is conservative for the BVPS-1 available NPSH calculation.

2. Enclosure 2 Section 3.1.1 (a) Verify that the table below is accurate for the utilization of analysis methods for the proposed containment conversion. (b) Describe the Modular Accident Analysis Program-Design Basis Accident (MAAP-DBA) long term (> 1 hour) Large Break LOCA (LBLOCA) energy release model. Reference the significant subroutines of MAAP 4.0, which are used for this calculation and describe any models and assumptions not already included in the MAAP 4.0 code. The description in this section states only that the enthalpy is obtained from MAAP-DBA. Is the mass release still obtained using the Westinghouse March 1979 model? (c) Describe the interfaces between the Westinghouse March 1979 model and MAAP-DBA for the different LOCA analyses and any iteration between these models for mass and energy and containment calculations.

DBA	Mass and Energy	Containment Response
LBLOCA (< 1 hour)	W March 1979	MAAP-DBA
LBLOCA (> 1 hour)	MAAP-DBA	MAAP-DBA
SBLOCA	MAAP-DBA	MAAP-DBA
MSLB	WCAP-8822;S1 and S2	MAAP-DBA

Response:

The above table is correct with the following clarification. The MAAP-DBA release model for greater than one hour utilizes the mass flow rates from the Westinghouse methodology. Furthermore, at this time into the accident, the mass flow discharged from the Reactor Coolant System (RCS) has reached equilibrium with the injection flow rate.

The long-term (greater than 3600 seconds) mass and energy release calculations are performed through user defined input functions by MAAP-DBA for the LBLOCA analyses. MAAP-DBA calculates the enthalpy of the release based on the enthalpy input from the sump water temperature as calculated by MAAP-DBA, the mass flow rate as characterized by the Westinghouse methodology and the several energy sources within the Reactor Coolant System (RCS). These energy sources include the core decay heat, the energy release rate from the RCS internals as the sump water cools below 212°F, and the energy release rate of the energy stored in the steam generator upper head internals and upper elliptical head (see Enclosure 2 Section 4.1.2). This approach is incorporated to provide a more accurate release enthalpy calculation since the Westinghouse methodology uses a fixed sump temperature assumption that does not incorporate the long term sump water cooling.

The interface between the Westinghouse March, 1979 model and MAAP-DBA is simply the timing assumed for the transition to the use of MAAP-DBA for the energy release calculation, i.e. one hour. For times less than one hour, the Westinghouse model is used to calculate the mass and energy releases. The Westinghouse model mass release history is used for times greater than one hour with the enthalpy calculated by MAAP-DBA as described above. An iterative process between the models for mass and energy and containment calculations has not been required given that the containment conditions in the mass and energy calculations are conservatively posed in the implementation of the Westinghouse March, 1979 model. Once the containment response is calculated with MAAP-DBA the conservatively assumed containment conditions are confirmed. The confirmation of the interface between these two models includes the ECCS recirculation time and recirculation temperature and the steam generator (SG) depressurization points (pressure and time). It is confirmed that the 1979 Model uses an earlier switchover time, a hotter recirculation temperature, and a quicker SG depressurization than predicted with MAAP-DBA.

3. Enclosure 2 Section 2: Do the Nuclear Steam Supply System (NSSS) parameters imply any restrictions on the fuel type, enrichment or nuclear operating strategy for either unit?

   Response:

   There are fuel type restrictions driven by the NSSS parameters. These restrictions would limit fuel re-designs to maintain the calculated core pressure drop and bypass flow which are inputs to the development of the NSSS parameters. Any change which would potentially impact these inputs and the associated parameters would require evaluations in many areas since the NSSS parameters are used to develop initial conditions for all safety analyses as well as containment analyses. There also exist specific fuel enrichment and power distribution limits, which are associated with source term development for radiological analysis. These are not directly associated with the NSSS parameters but are tracked as limits in the core reload design process. There are no restrictions on operating strategy associated with the NSSS parameters.

4. Enclosure 2 Section 3.2.2 Section 2.1 of WCAP-8822 identifies four break cases to be analyzed. Only two cases are described in these sections. Explain why the other two cases were not analyzed.

   Response:

   According to WCAP-8822 the following break areas are to be evaluated:

   1. A full double-ended (DE) rupture at the steam generator nozzle.
   2. A small DE rupture at the steam generator nozzle having an area just larger than that at which water entrainment occurs.
   3. A small DE rupture at the steam generator nozzle having an area just smaller than that at which water entrainment occurs.
   4. A small split rupture that will neither generate a steam line isolation signal from the Westinghouse Solid State Protection System (SSPS) nor result in water entrainment in the break effluent.

   Cases 1 and 4 were analyzed in the Beaver Valley Power Station main steamline break (MSLB) inside containment analysis as discussed in Enclosure 2 Section 3.2.2.

   Modeling water entrainment lowers the break release enthalpy, which in turn lowers the peak containment pressure. Case 1 assumed no entrainment and, therefore, bounds Cases 2 and 3.

5. TS Tables 3.3-3 for BVPS-1 and Unit 2: Containment Pressure Intermediate High-High Steamline isolation allowable value is given in the proposed TS change as 7.33 psig for BVPS-1 and 7.3 for BVPS-2. Is this correct?

Response:

Yes, BVPS-1 requires two decimal places and BVPS-2 requires only 1 decimal place. The difference is due to different instrumentation and specifically different instrument ranges.

Test instrumentation of both units has adequate accuracy and sensitivity to assure the analysis requirements are satisfied including allowances for uncertainties.

Test instrumentation has the capability to measure 3 decimal places. Thus, specifying 2 decimal places on BVPS-1 is well within the capability of the testing equipment.

The range of the BVPS-1 instruments is:-10.0 psig to 55 psig

The range of the BVPS-2 instruments is: -5.0 psig to 55 psig

The BVPS-1 Allowable Value (AV) is based on the calibration tolerance, which results in a two decimal AV. Rounding to one decimal reduces the AV and would unnecessarily restrict the calibration tolerance and AV.

The BVPS-2 AV is based on the calibration tolerance, for which the analysis conveniently results in the second decimal place being zero. The zero was dropped in rounding to a one decimal AV.

The same situation applies to the Containment High (SI actuation, FW isolation), Containment Intermediate High-High Phase A (CIA), and the Containment High-High Phase B (CIB) TS allowable values:

	BVPS-1	BVPS-2
Containment Press High	5.33 psig	5.3 psig
Containment Press Intermediate High-High	7.33 psig	7.3 psig
Containment Press High-High (CIB)	11.43 psig	11.4 psig

Analysis based on the unit specific instrumentation determined the above values.

6. Enclosure 2 Section 3.3 Provide the numerical result for a limiting case which demonstrates that the limited size of the break more than offsets the reduced RCS temperature for subcompartment analysis to verify the statement made in Section 1 of Enclosure 2 that there is no reduction in structural margin.

Response:

The current BVPS-2 peak calculated steam generator subcompartment wall pressure differentials were developed using a 33-node model following a 707 in^2 longitudinal intrados split break at the steam generator inlet elbow. The results are tabulated in Table 1 under the "Current Value" column heading (Ref. Beaver Valley Power Station Update Final Safety Analysis Report (UFSAR) Table 6.2-25). The mass and energy releases for this break were based upon an initial Reactor Coolant System pressure and temperature of 2,250 psia and 610.8°F, respectively (License Amendment Request (LAR), Enclosure 2, Table 2.1-1). The resulting pressure differentials were used in the structural analysis of the Steam Generator cubicle. Using the approved Leak-Before-Break (LBB) methodology, the maximum break size in the BVPS-1 and BVPS-2 steam generator subcompartment is limited to 14 inches and 6 inches in diameter, respectively.

Under Extended Power Uprate (EPU) conditions the corresponding initial Reactor Coolant System pressure remains 2,250 psia while the coolant system temperature varies between 603.9 to 617°F (LAR, Enclosure 2, Tables 2.1-2 and 3). The existing 33-node steam generator subcompartment model is employed with revised mass and energy release rates for the EPU conditions to demonstrate that the currently limited break size more than offsets any changes in Reactor Coolant System operating conditions. The break mass release rate are conservatively determined using the Westinghouse modified Zaloudek correlation (WCAP-8264) at 2,280 psia (includes 30 psi uncertainty) and 603.9°F and assumes no depressurization of the reactor vessel. In addition, the break effluent enthalpy is maximized assuming the highest operating temperature limit (i.e., 617°F at 2,220 psia including 30 psi uncertainty). The combination of high reservoir pressure with low temperature maximizes the mass release rate while using the enthalpy at the highest temperature maximizes the energy release rate. The resulting pressure differentials are tabulated in Table 1 under the "EPU Value" column. The current steam generator wall differential pressures used for the structural design are bounded for the EPU condition. This conclusion also applies to BVPS-1, since the hot leg double ended rupture break without LBB approach is the basis for the current steam generator subcompartment cubicle structural design.

The results demonstrate that if a maximum break size of 14 inch diameter is used in either Unit, the reduction in subcompartment wall differential pressures more than offsets the change in the reactor coolant operating temperature.

Table 1

BVPS-2 Steam Generator Subcompartment Peak Wall Differential Pressures

Node #	Current Power Value (psid)	Extended Power Uprate Value (psid)
1	15.08	5.73
2	10.50	4.07
3	10.34	5.14
4	10.66	5.86
5	10.44	6.10
6	9.86	5.37
7	12.32	6.54
8	10.62	4.07
9	10.68	4.05
10	10.25	4.44
11	10.08	4.83
12	9.96	3.38
13	9.92	4.45
14	9.85	3.71
15	10.52	4.36
16	9.62	3.82
17	9.43	3.82
18	9.29	3.47
19	9.34	3.53
20	9.16	3.53
21	9.09	3.23
22	9.54	3.36
23	9.19	3.46
24	9.13	3.61
25	1.50	0.74
25	1.37	0.70
27	1.09	0.54
28	1.01	0.55
29	8.18	2.67
31	13.27	7.15
32	26.94	13.78
33	12.66	6.92

Note that the BVPS-2 UFSAR Table 6.2-25 does not list the results for node 32. The value was added for completeness.

7. Enclosure 2 Section 1.3: An evaluation of breaks in the BVPS-1 main steam valve room is added to the BVPS-1 licensing basis to assess the impact of adding cavitating venturis to the auxiliary feedwater lines. Is there already such an analysis for BVPS-2? Is it affected by EPU application dated October 4, 2004?

   Response:

   Although not described in the containment conversion submittal, a re-analysis of the BVPS-2 Main Steam Valve House (MSVH) environment has been performed to address the impacts of EPU. This analysis includes the effect of cavitating venturis since these were existing at BVPS-2. The BVPS-2 analysis is discussed in Enclosure 2 Section 10.10 of LAR 302/173, Extended Power Uprate, and concludes that the equipment in the affected areas is either not required to mitigate the accident or have performed the intended safety function prior to being challenged by the peak accident temperatures. The BVPS-1 re-analysis, described in Enclosure 2 of the containment conversion LAR, includes the effect of both the EPU and installation of the cavitating venturis.

8. Enclosure 2 Section 3.1.3.4 Which single failure is limiting for the LOCA mass and energy release? If the loss of a train of service water is limiting, why shouldn't it be considered for BVPS-1 also?

   Response:

   There is no limiting single failure for all analyses. The limiting single failure depends upon the attribute being evaluated (e.g., containment peak pressure, NPSH, etc.). In general, for LOCA mass and energy release calculations, only two conditions are evaluated. The first condition is the loss of an emergency diesel generator, which results in minimum safeguards which is the loss of one train of all safeguards equipment. The second condition is maximum safeguards. For maximum safeguards cases, no failures are postulated to occur which affect the safety injection system or the mass and energy release calculations although other failures may be postulated when this data is utilized in the containment analysis (e.g. quench spray pump failure, or a train of service water).

   At BVPS-2, a portion of the recirculation spray system is utilized to provide low head safety injection flow in the recirculation mode of safety injection. The water from the containment sump is pumped through a recirculation spray heat exchanger to the low head safety injection header and the suction supply piping to the high head safety injection pumps. The heat exchangers are cooled by service water. In order to establish the maximum temperature of the water pumped into the Low Head Safety Injection (LHSI) piping for thermal stress considerations, a case is run which assumes a single failure of the service water supply to the recirculation spray heat exchangers. This failure is not applicable to BVPS-1 since the recirculation spray pumps are not used to provide LHSI flow.

9. Enclosure 2 Section 3.1.6 (a) What are the characteristics of the heat exchanger used for NPSH calculations? (b) What service water temperature is assumed? (c) Describe the program to ensure that the heat exchanger characteristics assumed in the analyses are not exceeded?

Response:

The recirculation spray heat exchangers are the only heat exchangers used to remove heat from the containment following a CIB signal. The characteristics of the heat exchangers represent a range of performance conditions and the characteristics used for a particular NPSH analysis are the most conservative as determined by sensitivity studies. For BVPS-1, the NPSH analyses are sensitive to the heat exchanger performance since the characteristics influence both the sump water temperature and containment pressure. These parameters are both direct inputs into the NPSH calculation for Recirculation Spray (RS) and LHSI pumps since containment overpressure is credited. For BVPS-2, the NPSH calculation is insensitive to heat exchanger characteristics since the NPSH calculation is dependent only on sump level and RS pump flow (BVPS-2 uses the saturated sump model).

Maximum performance characteristics of the heat exchangers are limiting for BVPS-1 RS pumps NPSH. The characteristic for maximum performance are based on no tubes plugged, and zero fouling conditions. This case also uses the minimum service water temperature. This is most limiting since maximum heat removal minimizes the RS temperature which causes the highest containment de-pressurization rate. Since containment pressure is a component of the NPSH calculation, minimizing the pressure lowers the available NPSH. While maximizing heat removal also ultimately lowers containment sump temperature as compared to minimum heat removal, this occurs over a much longer time basis. The transient effect of lowering containment pressure faster, relative to sump temperature following RS pump startup results in minimum available NPSH for the RS pumps for maximum heat removal conditions.

For the BVPS-1 LHSI pumps NPSH, minimum heat removal conditions are limiting. This results in the use of minimum performance characteristics for the heat exchangers, i.e., maximum allowed tube plugging and design heat exchanger fouling levels. This case also uses maximum service water temperature. The LHSI pump NPSH analysis credits containment overpressure. Minimum available NPSH for these pumps occurs at switchover to recirculation mode when the pumps begin to initially draw from the containment sump. Since the containment has substantially de-pressurized by the time switchover occurs, the contribution and variability of containment pressure to the NPSH calculation is low. The available NPSH is much more sensitive at this point to containment sump temperature (and resulting vapor pressure). Therefore, conditions which maximize sump temperature at switchover, tend to minimize available NPSH. For this reason, minimum heat removal via the RS heat exchangers has been shown to be limiting for this case.

The heat exchanger monitoring program at both units requires that these heat exchangers are mechanically cleaned and inspected on a regular frequency. The inspection includes visual examination and eddy current testing of the tube integrity. The tube plugging limits are controlled to be within the assumptions in the performance analysis. This method of ensuring recirculation spray heat exchanger heat transfer capability has been accepted by the NRC as an equally effective alternative to testing that satisfies Generic Letter 89-13, SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY- RELATED EQUIPMENT, SUPPLEMENT.

10. Enclosure 2 Section 3.1.5.7 It was assumed that no zirconium-water (Zr-water) reaction would occur. Please discuss the basis for this assumption. Does the design-basis LOCA calculation done in accordance with Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Section 50.46, for BVPS-1 or -2 show a Zr-water reaction?

    Response:

    Enclosure 2 contains the results and conclusions for the LBLOCA mass and energy release calculations. Section 3.1.5.7, "Sources of Mass and Energy", of Enclosure 2 states:

    "The zirc-water reaction energy was not considered in the mass and energy release data presented because the clad temperature was not assumed to increase high enough for the rate of the zirc-water reaction to be of any significance."

    The energy release from the Zr-water reaction was not considered as part of the WCAP-8264-P-A, Rev.1 or the WCAP-10325-P-A methodology. Based on the manner in which the energy in the fuel is conservatively modeled to be released to the vessel fluid, the fuel cladding temperature does not increase to the point were the Zr-water reaction is significant. This is in contrast to the 10CFR50.46 analyses, which are biased to calculate high fuel rod cladding temperatures, which predicts a significant Zr-water reaction. For the LOCA mass and energy calculation, the energy created by the Zr-water reaction value is small and is not explicitly provided in the energy balance tables. The energy that is determined is part of the mass and energy releases and is therefore already included in the LOCA mass and energy release calculations presented in Enclosure 2.

11. Updated Final Safety Analysis Report (UFSAR), BVPS-1, Table 14.3-16: This table shows that the pump discharge double ended rupture (DER) gives the highest peak pressure for the current power level. Why wasn't this case analyzed for the containment conversion? Also the current pump discharge DER analysis yields a higher pressure than the hot leg DER. Explain why this is contrary to the containment conversion analysis.

    Response:

    The BVPS-1 limiting cases changed from those currently presented in the BVPS-1 UFSAR because the BVPS-1 mass and energy methodology is being changed from LOCTIC to the Westinghouse mass and energy release methodology. For BVPS-2, the Westinghouse mass and energy release methodology is currently used, thus the limiting cases are unchanged from what is currently presented in the BVPS-2 UFSAR.

    The results in the current BVPS-1 UFSAR are based on LOCTIC analysis. For BVPS-1, the LOCTIC program is used for both mass and energy release calculation as well as the containment response. The LOCTIC program is designed to be conservative with respect to peak containment pressure and is not utilized to perform core accident analysis. The program calculates the flow of steam, water, or a two-phase mixture from the reactor vessel through the piping to and out the break. The flow rate is dependent upon break location and size, fluid flow friction coefficients, liquid level in reactor, containment pressure, and state of primary coolant after heat and mass exchanges from hot metal sources, decay heat production, sensible heat from the core, power coastdown heat, zirc-water reactions, safety injection and reactor vessel break flow. Since the LOCTIC program employs conservative critical flow correlations and other conservative input parameters such as low fluid flow friction coefficients and high heat transfer coefficients, the mass and energy release rates determined during the blowdown phase are conservative for the containment analysis. The current results using LOCTIC show that the pump discharge DER containment peak pressure is slightly higher than the hot leg DER by 0.09 psig, which is mainly due to the conservatively low fluid flow friction coefficient assumed across the reactor coolant pump for the pump discharge DER case (i.e., no locked rotor case).

Enclosure 2 reflects the results of Westinghouse mass and energy release methodology for both BVPS units. As shown below, extracted from Table 6.2-4 of the BVPS-2 UFSAR, the Hot Leg DER break is limiting.

- Hot Leg DER (DEHL) Peak Calculated Pressure = 44.66 psig
- RCS Pump Discharge (DECL) Peak Calculated Pressure = 42.22 psig
- RCS Pump Suction (DEPS) Peak Calculated Pressure = 42.21 psig

The peak pressure occurs during blowdown and the cold leg (RCS pump discharge) break is bounded by the DEHL break during blowdown. There is very little difference during blowdown between the DECL and the DEPS break, however both are bounded by the DEHL break.

The BVPS-2 results are consistent with the current Westinghouse methodology. Based on studies performed with more detailed Westinghouse methodology, the cold leg break (pump discharge break) location is less limiting than other break locations (hot leg and pump suction breaks). This is discussed in Enclosure 2 Section 3.1.3.3 of the LAR. Since the analysis supporting this LAR uses the Westinghouse methodology, the cold leg break location was not analyzed.

12. Enclosure 2: Provide the peak containment atmosphere temperature for the different LOCA cases in the Tables 4-16 and 4-17.

Response:

The peak containment atmosphere temperature for the different LOCA cases in Tables 4-16 and 4-17 are presented below. The containment peak pressure results presented in Tables 4-16 and 4-17 were biased to produce the peak pressure response. Consistent with a design basis approach, the results summarized below for peak containment atmosphere temperature were based on inputs biased to maximize the containment temperature.

Design Basis Large Break LOCA Beaver Valley Power Station - BVPS-1

LOCA Case	Peak Temperature °(F)
6L-DEPS MIN SI	266.0
7L-DEPS MAX SI	266.0
8L-DEHL	267.8

Design Basis Large Break LOCA Beaver Valley Power Station - BVPS-2

LOCA Case	Peak Temperature °(F)
1L-DEPS MIN SI	266.7
2L-DEPS MAX SI	266.6
3L-DEHL	270.1

13. Specify which variables have been assigned uncertainties for the mass and energy and containment calculations and the values of those uncertainties for LBLOCA, Small Break LOCA (SBLOCA), and Main Steamline Break (MSLB). Enclosure 2 Section 3.2.2.1 states that uncertainties are only applied to those parameters that could increase the amount of mass or energy discharged into containment. Is this a change from the method described in WCAP-8822?

Response:

The LBLOCA mass and energy releases were calculated using the methodology found in WCAP-10325-P-A and WCAP-8264-P-A, Rev.1. The following describes the variables that have had uncertainties applied in order to assure that LBLOCA mass and energy releases have been maximized.

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the Reactor Coolant System (RCS) operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of (+4.0°F on BVPS-1 and +4.0°F on BVPS-2) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+40 psi on BVPS-1 and + 42 psi BVPS-2).

All input parameters are chosen consistent with accepted analysis methodology. Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed next. Tables 3.1-1 and 3.1-28 of Enclosure 2, present key data assumed in the analysis.

The core rated thermal power of 2900 MWt adjusted for calorimetric error (+0.6 percent of power) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS to the temperatures. The selection of 2290 psia for BVPS-1 and 2292 psia for BVPS-2 as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term M&E release calculations.

The selection of the fuel design features for the long-term Mass and Energy (M&E) release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The margin in core-stored energy was chosen to be +15 percent for Unit 1 and a statistical uncertainty was used for BVPS-2. Thus, the analysis for both units very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3 percent (1.6 percent allowance for thermal expansion and 1.4 percent for uncertainty) is modeled.

A uniform steam generator tube plugging (SGTP) level of zero percent (0%) is modeled. This assumption maximizes the reactor coolant volume and fluid release by considering the RCS fluid in all SG tubes. During the post-blowdown period the steam generators are active heat sources, as significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The zero percent SGTP assumption maximizes heat transfer area and therefore, the transfer of secondary heat across the SG tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the pressure drop upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the level of SGTP

Regarding safety injection flow, the M&E release calculation considered configurations/failures to conservatively bound respective alignments. These cases include (1) a Minimum Safeguards case (one Charging/Safety Injection pump [CH/SI] and one Low Head Safety Injection [LHSI] pump) and (2) a Maximum Safeguards case (two CH/SI and two LHSI pumps). (Also see the response to question 8.)

The following assumptions were employed to ensure that the M&E releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS (100-percent full-power conditions)
2. Allowance for RCS temperature uncertainty (+4.0°F)
3. Margin in RCS volume of 3 percent (which is composed of 1.6-percent allowance for thermal expansion, and 1.4 percent for uncertainty)
4. Core rated power of 2900 Megawatts thermal (MWt).
5. Allowance for calorimetric error (+0.6 percent of power)
6. Conservative heat transfer coefficients (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)
7. Allowance in core-stored energy for effect of fuel densification
8. A margin in core-stored energy (+15 percent for Unit 1 and Statistical for BVPS-2)
9. An allowance for RCS initial pressure uncertainty (+40psi on BVPS-1 and +42psi on BVPS-2)
10. A maximum containment backpressure equal to design pressure (45 psig)
11. SGTP leveling (0-percent uniform)
    - Maximizes reactor coolant volume and fluid release
    - Maximizes heat transfer area across the Steam Generator (SG) tubes
    - Reduces coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow.

Thus, based on the previously discussed conditions and assumptions, a bounding analysis for BVPS-1 and -2 was made for the release of M&E from the RCS in the event of a LOCA at 2900 MWt.

For the MSLB the following information is provided:

a.) Only critical parameters have been assigned uncertainties for the MSLB mass and energy calculations:

- Power Fraction (+0.6%)
- RCS Average Temperature (+8.5 °F)
- Steam Generator Water Level (7 % Narrow Range Span (NRS))

The method of WCAP-8822 describes how to analyze the "worst case" scenario for the MSLB event, indicating which parameters are critical to increasing the mass and energy releases inside containment. The use of uncertainties for the initial conditions associated with full power, RCS average temperature, and steam generator water level are consistent with the conservative intent of the method documented in WCAP-8822. Of these parameters, the only one for which an uncertainty was applied is the full-power initial condition. All other process parameters used in the analysis documented in WCAP-8822 assumed nominal or design values for the initial conditions. The use of uncertainties for the RCS average temperature and the steam generator water level for Beaver Valley Power Station provide conservative initial conditions for the plant-specific analysis. Thus, there is no change from the method described in WCAP-8822 used for the Beaver Valley Power Station analyses.

The SBLOCA mass and energy release calculation assigned uncertainties to the controlling input parameters of initial core power, decay heat, and the break discharge coefficient. The core rated thermal power of 2900 MWt adjusted for calorimetric error (+0.6 percent of power) was used in the analysis. The decay heat curve used the ANSI/ANS 5.1 1979 data with 2-sigma uncertainty. The break discharge coefficient value was set at 1.0. The range of small break LOCA sized was extended to include intermediate break sizes up to 12 inch diameter in both the cold and hot leg. This extensive range of break sizes (1 inch through 12 inch) and locations provides a means of addressing uncertainties in the mass and energy releases.

The containment calculations are also addressed in the response to RAI item 1 and Table 4-3 of Enclosure 2. This table identified values used in calculations and summarizes the manner in which the 24 initial conditions and parameter values were selected to conservatively bias the containment accident peak pressure, NPSH, peak equipment qualification temperature, peak liner temperature, maximum sump water temperature and maximum service water outlet temperature.

14. <u>Enclosure 2 Section 4.2.1</u>: (a) Describes the modeling of the droplets formed by 10% of the nonflashed break flow. Include entrainment, gravitational settling, impaction on solid surfaces, coalescence, vaporization and condensation. (b) What is the basis for the assumed 100 micron droplet size for break flow? (c) What is the basis for the 10% fraction of the non-flashed flow which is droplets?

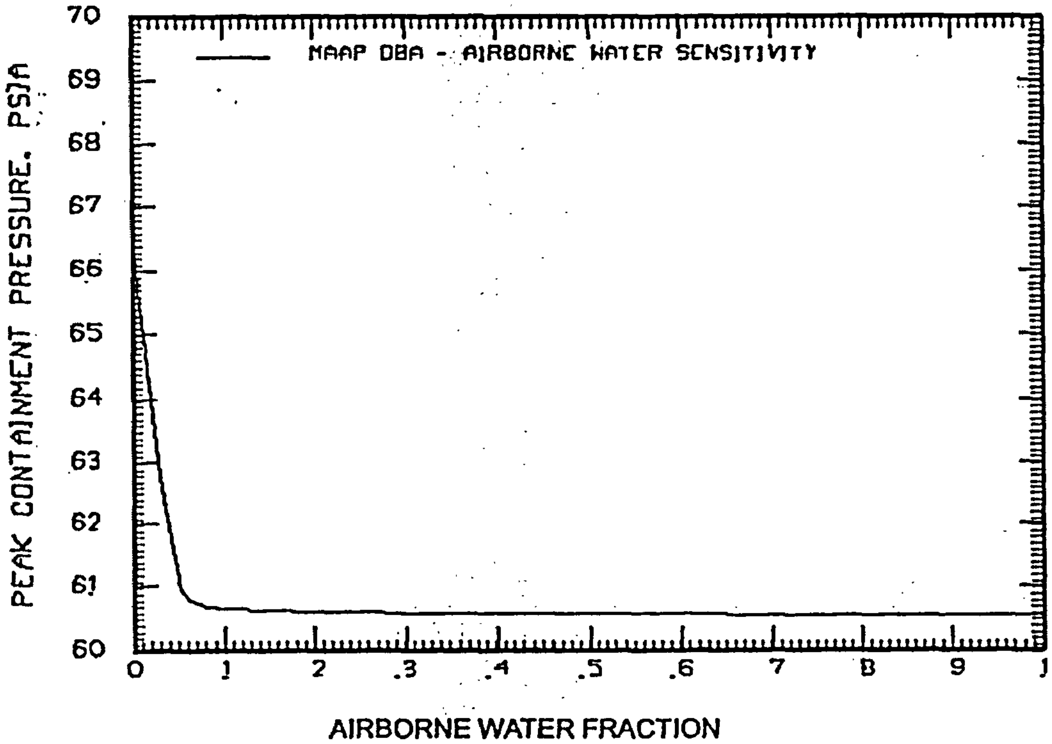
Response:

MAAP-DBA credits water droplets originating in the blowdown jet into the containment atmosphere following LOCA. During and following a blowdown transient, airborne water acts as an internal distributed heat sink for steam to minimize superheat as well as to condense some of the steam mass. With the large surface-to-volume ratio, combined with high water density and specific heat compared to steam, a relatively small droplet water mass is sufficient to eliminate steam superheat from the containment atmosphere. During the blowdown, the fraction of available liquid airborne is specified to be 10% with a droplet size of 100 microns. Following blowdown (about 30 seconds or less) the fraction of available liquid airborne is specified as 0%, and the remaining airborne water mass decreases due to deposition, potentially spray operation, or evaporation. The deposition is calculated with the models in the FPTRAN sub-routine as described in the MAAP4 User's Manual. The deposition mechanisms addressed in this sub-routine include sedimentation, impaction, containment leakage, diffusiophoresis, and thermophoresis. Evaporation is calculated with the models described in Section 2.1.1.6 of the pre-submittal report (FENOC Letter L-03-188, L. W. Pearce dated November 24, 2003, Subject: Beaver Valley Power Station, Unit 1 and 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73 Containment Conversation Pre-Application Report).

The droplets generated at the break during blowdown will be on the order of 100 microns or less based on measurements for discharge of superheated (relative to the discharge environment) water. Other containment analysis codes such as GOTHIC also use this drop size. GOTHIC's quantification analyses performed using a drop size of 100 microns were found to agree with and typically bound the measured pressure and temperature response for blowdown tests and measured pressure drops from orifice pressure drop tests.

The bases for the 10% fraction of the nonflash flow which is droplets include integral effects containment experiments and sensitivity analyses. The CASP1 (D15) and CASP2 (D16) experiments performed in the Battelle-Frankfurt model containment (as discussed in 9.4.3) show the aerosolization and airborne transport of water during blowdown. Assessments of the results for these two tests indicate values of 49% and 65% for the water masses transported as liquid outside of the break room. Consequently, these experiments undergoing a high pressure two-phase critical discharge clearly have a large fraction of water transported away from the break room. It is also noted that these experiments were performed with an impingement (baffle) plate immediately downstream of the break, which acts to strip-out (reduce) the airborne water mass. These results demonstrate that the assumed airborne fraction of 10% used in the BVPS-1 and -2 containment conversion analysis is a conservatively low value. Additionally, the pre-submittal report (FENOC Letter L-03-188) provided by FENOC presented a sensitivity of containment pressure to LOCA airborne water fraction using the MAAP-DBA code and the single node model. The MAAP-DBA code single node model using the Tagami heat transfer correlation was run for a series of assumed airborne water fractions. For airborne water fractions of 0.1 to 1 the calculated peak containment pressure was shown to be insensitive as shown in the following figure.

Sensitivity of Containment Pressure to LOCA Airborne Water Fraction



15. UFSAR, BVPS-1 Section 6.3.3.9 (a) The BVPS-1 FSAR states that the current NPSH calculations are done assuming the pressure flash model for the break flow. Which is more conservative, the pressure flash model or the break mixing model? If the pressure flash model is more conservative, justify using the break mixing model. (b) Is the break droplet size equivalent to a spray efficiency of 100%?

Response:

The pressure flash model and the break mixing model address related but different aspects of the NPSH calculations. MAAP-DBA uses a pressure flash model to assess RCS blowdown. For NPSH transient analyses it is desirable to maximize energy addition to the containment sump since this maximizes the sump water temperature while minimizing the containment pressure. This was done in the UFSAR BVPS-1 analysis (Section 6.3.3.9) in the current available NPSH assessment. This same conservatism was also implemented in the MAAP-DBA assessment of the available NPSH. The break mixing model was used to accomplish this conservatism by mixing the two streams being discharged from the postulated double ended break to assess the break enthalpy. The pressure flash model is then applied to the mixed stream to determine the amount of break fluid that is directed as saturated water to the containment sump or steam to the containment atmosphere. This procedure increases the amount and enthalpy of water directed to the sump compared to that which would result from not using the mixing model.

The MAAP-DBA model is applied for the NPSH calculations such that droplets from the break are only considered during the initial blowdown, i.e. approximately a 20 second interval. Subsequent to that time none of the liquid break flow is considered to become an airborne droplet. The treatment of the break droplet size of 100 microns during this initial phase of the calculation is equivalent to a spray efficiency of 100% in that the droplets come into thermal equilibrium with the containment atmosphere during their residence time.

16. UFSAR BVPS-1 Section 6.3.3.9: Verify that the limiting NPSH for the safety injection charging pumps still occurs at the end of the injection phase.

Response:

As shown in Table 4, the minimum NPSH available for the high head safety injection charging pumps is 46.70 feet for BVPS-1 and 43.90 feet for BVPS-2. This minimum available NPSH occurs at the end of the injection phase of safety injection (while the suction supply to the pumps is still from the Refueling Water Storage Tank (RWST)). During the recirculation phase of safety injection, the minimum available NPSH for the pumps is 52 feet for BVPS-1 and 59 feet for BVPS-2. Therefore, the limiting NPSH occurs for the high head safety injection charging pumps at the end of the injection phase.

17. Verify that all parameters covered by technical specifications are at the conservative technical specifications limit for the mass and energy and containment calculations.

Response:

LOCA M&E Analysis Parameter/ Value	SLB M&E Analysis Parameter/ Value	Containment Analysis Parameter/ Value	Current technical specification Value	Containment Conversion LAR Nos. 317 and 190 Value
	Lead Time Constant for Steamline Pressure Low 50 seconds Lag Time Constant for Steamline Pressure- Low 5 seconds		Not contained in BVPS-1 technical specifications $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds for Functions 1. e and 4. d in Table 3.3-3 of the BVPS-2 technical specifications	
Accumulator Volume 7,645 (maximum useable)(2) gallons for BVPS-1 Accumulator Volume 8019 (maximum useable)(1) gallons for BVPS-2			Between 7,664 and 7,816 gallons (contained)(2) in LCO 3.5.1 b. for BVPS-1 Between 7,532 and 7,802 gallons (contained)(1) in LCO 3.5.1 b. for BVPS-2	
Accumulator Pressure 560 psig (minimum)(3)			Between 605 and 661 psig in LCO 3.5.1 d. for BVPS-1(4) Between 585 and 665 psig in LCO 3.5.1 d. for BVPS-2(3)	
		Containment Internal Pressure Between 12.8 and 14.2 psia		Containment Internal Pressure -$\geq$ 12.8 and $\leq$ 14.2 psia in technical specifications 3.6.1.4 for BVPS-1 and -2
Containment Air Temperature-105°F		Containment Air Temperature-70-105°F		Containment Air Temperature-$\geq$ 70°F and $\leq$ 105°F in technical specifications 3.6.1.5 for BVPS-1 and -2
Diesel Generator Start Time- 12 seconds			Diesel Generator Start Time- $\leq$ 10 seconds in SR 4.8.1.1.2 b. 3. b) for BVPS-1 and -2	

Notes:

(1) The BVPS-2 technical specification maximum contained volume of 7,802 gallons corresponds to a maximum useable volume of 7,604.5 gallons, which is less than the analysis maximum useable volume of 8,019 gallons. Therefore, the analysis uses a conservatively high maximum usable volume, which is conservative.

(2) The BVPS-1 technical specification maximum contained volume of 7,816 gallons corresponds to a maximum useable volume of 7,601.2 gallons, which is less than the analysis maximum usable volume of 7,645 gallons. Therefore, the analysis uses a conservatively high maximum usable volume, which is conservative.

(3) The BVPS-2 technical specification minimum pressure of 585 psig is greater than the analysis minimum pressure of 560 psig. Therefore, the analysis uses a conservatively low minimum pressure, which is conservative.

(4) The BVPS-1 technical specification minimum pressure of 605 psig is greater than the analysis minimum pressure of 560 psig. Therefore, the analysis uses a conservatively low minimum pressure, which is conservative.

The Reactor Trip safety analysis limit trip setpoint for Pressurizer Pressure- Low and ESFAS safety analysis limit trip setpoints for Safety Injection and Feedwater Isolation on Pressurizer Pressure- Low and Steamline Pressure Low, and Steam Line Isolation on Steamline Pressure-Low assumed in the LOCA and SLB Mass and Energy Release analyses, are considered in the determination of the Nominal Trip Setpoints (NTS) which are contained in Tables 3.9-1 and 3.9-2 of the Licensing Requirements Manual (LRM), and include the appropriate instrument uncertainties. Technical specification 3.3.1.1, "Reactor Trip System Instrumentation," and 3.3.2.1, "Engineered Safety Features Actuation System Instrumentation," contain the Allowable Values associated with these Nominal Trip Setpoints.

The response times for Reactor Trip (from SI), Safety Injection, Feedwater Isolation, and Steam Line Isolation assumed in the SLB Mass and Energy Release analyses are contained in Tables 3.1-1 and 3.2-1 of the LRM.

The values for $F_Q(Z)$ and $F^N_{\Delta H}$ assumed in the LOCA Mass and Energy Release analyses bound those contained in the Core Operating Limits Report (COLR). These values are reviewed prior to each fuel cycle to confirm that the values assumed in the safety analyses remain bounding or equivalent.

18. Does the NPSH calculation assume maximum engineered safety features are operable as specified in Section 6.4.3 of the BVPS-1 FSAR?

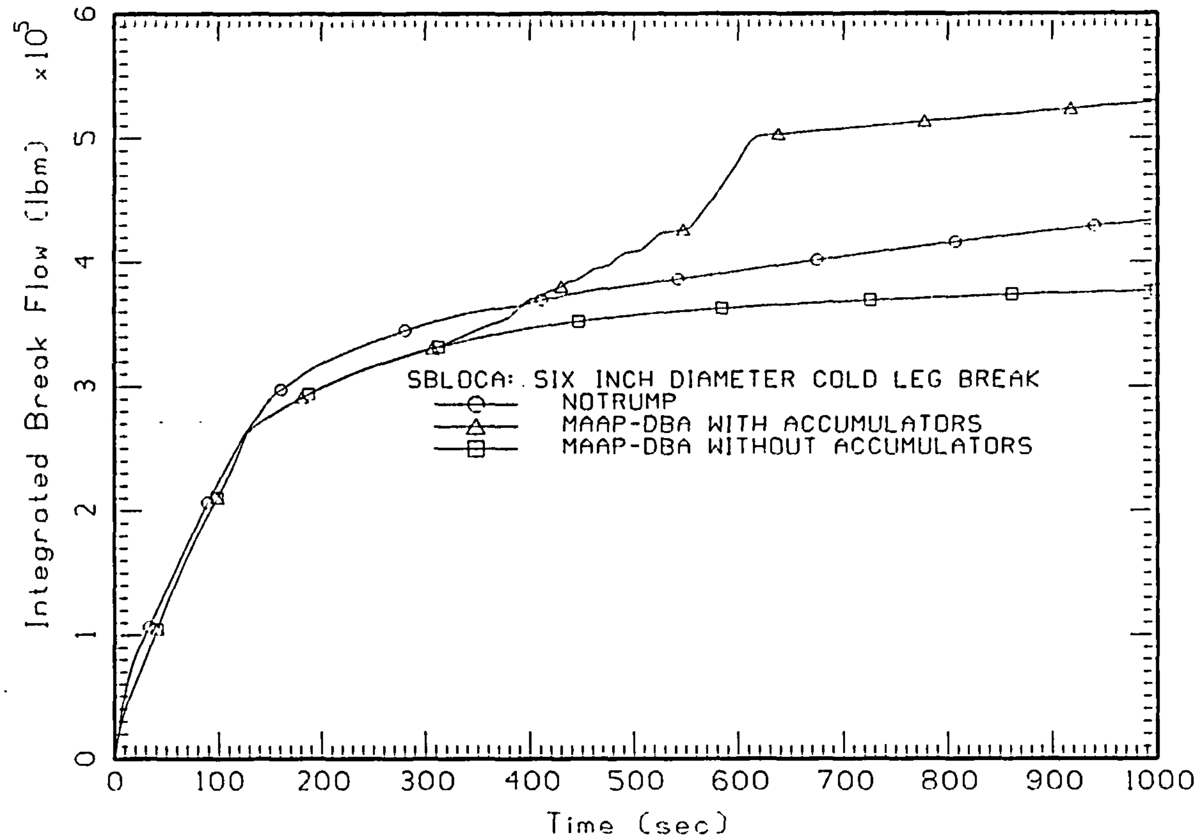
Response:

Yes, the available NPSH calculations include cases assuming maximum engineering safety features are operable as specified in Section 6.4.3 of the BVPS-1 UFSAR. The available NPSH calculations performed with MAAP-DBA considered various single failures, break locations, and ranges of initial conditions and performance parameters to determine the limiting case for the available NPSH for the RS pumps. The run matrix (see Enclousure 2 Table 4-1) and the parameter matrix (see Enclousure 2 Table 4-3) summarize the spectrum of cases and conditions used for the available NPSH calculations.

19. Enclosure 2 Figure 9-29a: Explain why the two curves shown in the figure diverge significantly beginning around 400 seconds.

Response:

The two curves shown in Enclosure 2 Figure 9-29a indicated the integrated break flow (lbm) for the MAAP-DBA and NOTRUMP 6 inch diameter cold leg break. The two curves diverge significantly beginning around 400 seconds due to the difference in the simulated performance of the accumulators and the impact of the injected water on the primary system conditions. The MAAP simulation indicates a higher integrated mass flow rate since the accumulator is calculated to have completely discharged its inventory into the primary system. By approximately 825 seconds the accumulator is completely depleted in the MAAP-DBA simulation and partially depleted in the NOTRUMP simulation. The attached figure includes a third curve, which was calculated with MAAP-DBA with the accumulators disabled. The two sets of MAAP-DBA results bracket the NOTRUMP prediction and illustrate the impact of completely depleting the accumulator inventories in the MAAP-DBA prediction.



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20. Enclosure 2 Section 9.5: (a) Describe the MAAP-DBA model used for the small break mass and energy release calculations. (b) Provide a noding diagram for the MAAP-DBA small break analyses. (c) Describe the break flow model used and the C_D value used. (d) Provide results of coolant level vs. time for the 2-inch and 6-inch Beaver Valley-specific small break calculations. (e) Explain why, although the MAAP- DBA model will be used for breaks up to 12 inches, no benchmarks are done for breaks greater than 6 inches. (f) MAAP-DBA is benchmarked against NOTRUMP for SBLOCA mass and energy release. Neither code has the capability to model multi-dimensional and entrainment effects (e.g., emergency core cooling bypass and asymmetric downcomer boiling). Please explain why it is acceptable to neglect these effects when calculating break flow for the larger small breaks.

Response:

Part (a) - A simplified version of the Henry-Fauske two-phase critical flow model is used for the small break mass and energy release calculations. A description of this model is provided in the MAAP4 User's Manual as part of subroutine WFLOW.

Part (b) - Figure 1 provides the noding diagram for the MAAP-DBA LOCA analyses. Note that the loop with the pressurizer is designated as the "broken loop" which has experienced the postulated pipe break, while the "unbroken loop" represents the two loops that are not postulated to have a pipe break.

Part (c) - The break flow model uses a simplified version of the Henry-Fauske two-phase critical flow model and a value of $C_D = 1$.

Part (d) - Figure 2 illustrates the MAAP-DBA calculated water levels in the RPV downcomer for the 2-inch and 6-inch cold leg breaks. The water level is referenced to the elevation of the bottom of the RPV, and the break elevation in the cold leg is 26.6 ft.

Part (e) - Benchmark evaluations of the MAAP-DBA mass and energy release model was focused on the range of break sizes (2 to 6-inch diameter) that were found to be limiting for the minimum sump inventory (MSI) calculations (see Table 2), which addressed one of the key plant design attributes. The recirculation spray pump available NPSH and MSI are the only plant design attributes that are not set by a large break LOCA or MSLB. A small and intermediate break LOCA assessment was conducted to confirm that the large LOCA and MSLB sequences produced the design basis limiting results as part of the containment conversion calculations. The small break LOCA sizes were extended to include the intermediate break sizes up to 12 inch diameter for this demonstration. This extensive range of break sizes provides a convenient means of addressing uncertainties in the calculated mass and energy releases, i.e. break geometry, break discharge coefficient, etc. Furthermore, the MAAP-DBA model benchmarking provided in the containment conversion submittal has been expanded by comparison to a 12-inch line bottom vessel blowdown experiment, i.e. Test No. 6 of MARVIKEN critical flow test series. The results from the comparison between the experimental measurements and the MAAP-DBA prediction (using $C_D = 1.0$) are presented in Figures 3 and 4. The benchmark demonstrates the suitability of applying the MAAP-DBA critical flow model for blowdown for larger break sizes since it provides conservative prediction of the mass discharge rate.

Part (f) - Multiple-dimensional and entrainment effects may be viewed as being of potential importance regarding the quantification of mass and energy releases for larger small breaks, i.e. the intermediate range of break sizes. These effects could impact the mass and releases for cold leg break locations but are not expected to have a significant impact on hot leg breaks of any dimension. Thus, the acceptability of neglecting these effects in the MAAP-DBA calculations should be focused on larger cold leg breaks. The bounding results from the small and intermediate break study for the Beaver Valley Power Station that uses MAAP-DBA to quantify the mass and energy release histories have been reviewed to identify the limiting case for each design basis attribute for each unit. Except for the minimum sump inventory (MSI) attribute, hot leg break results are found to bound cold leg break results. The bounding results for small and intermediate break sizes (1 to 12 inch diameter) are compared to the large LOCA and MSLB results to see if any of the small or intermediate size breaks establish the design basis value. The Table 3 provides a comparison for both BVPS-1 and BVPS-2 of the design basis values versus the bounding results for hot leg and cold leg breaks studied in the Beaver Valley Power Station small/intermediate break assessment. Table 3 indicates that the design basis values were set by the results from double-ended hot leg breaks, double-ended pump suction breaks, or main steam line breaks, for all attributes with the exceptions of the minimum sump inventory (MSI) attribute and the available NPSH for the recirculation spray. For these exceptions the results from small (3 inch hot leg), (4 and 5 inch cold leg), and a large (12 inch hot leg) diameter breaks were bounding. A hot leg break is not expected to be impacted by these effects. Likewise, due to their small size these 4 and 5 inch diameter cold leg breaks are considered to be negligibly impacted by multi-dimensional and entrainment effects. Furthermore, should these effects impact the mass and energy release it is expected that the inventory released by the break to the containment would be enhanced. For the minimum sump inventory and available NPSH assessments it is conservative to minimize the containment sump water inventory, i.e. it is conservative for the RCS to retain the maximum coolant inventory. Hence, for these attributes it is concluded that ignoring these effects is conservative should they impact the RCS release. Thus it is concluded that the bounding small LOCA results calculated by MAAP-DBA are conservative and acceptable since these multiple-dimensional and entrainment effects are neglected.

Table 2

Minimum Sump Inventory (Water Level) Results

Break Size (Inch)	Minimum Sump Water Level (ft)	
	BVPS-1	BVPS-2
1	2.21	2.54
2	2.10	2.51
3	2.04	2.42
4	2.02	2.40
5	2.03	2.38
6	2.23	2.87
7	2.24	2.89
8	2.25	2.94
9	2.27	3.00
10	2.29	3.06
11	2.33	3.11
12	2.36	3.16

Table 3

Comparison of Design Basis Limiting Results and Small/Intermediate Break Results

				MST		NPSH				
	PP (psig)	PT (F)	MLT (F)	RS (F)	ECCS (F)	IRS/RS (ft)	ORS (ft)	LHSI (ft)	MSI (ft)	MSW (F)
BVPS-1										
DB	43.3 DEHL/NF(1)	355 MSLB/MSCV	254.1 MSLB/MSCV	244 DEPS/ MIN SI/DG	183.5 DEPS/ MIN SI/DG	15.1 DEHL/ MAX SI/QS	12.27 12 in/QS	27.17 DEPS/ MIN SI/DG	2.02 4 in/QS	196 DEPS/ MAX SI/CIB
HL	29.9 12 in/DG	260.74 9 in/NF	217.81 12 in/DG	231.8 12 in/QS	150.45 8 in/DG	18.07 12 in/QS	12.27 12 in/QS	30.7 12 in/DG	2.534 6 in/DG	182.13 12 in/QS
CL	27.17 12 in/QS	256.53 12 in/QS	214.6 12 in/DG	201.48 5 in/QS	145.45 12 in/DG	30.31 12 in/NF	25.02 12 in/QS	31.34 12 in/QS	2.02 4 in/QS	159.07 5 in/QS
BVPS-2										
DB	44.9 DEHL/NF	332 MSLB/MFIV	247.7 MSLB/MSIV/ CIB	246 DEPS/ MIN SI/DG	169.3 DEHL/CIB	15.1 3 in/DG	N/A	N/A	2.38 5 in/QS	175 DEPS MIN SI/DG
HL	30.15 12 in/DG	255.2 12 in/DG	216.85 12 in/QS	228.6 12 in/DG	135.4 12 in/DG	15.1 3 in/DG	N/A	N/A	2.61 3 in/DG	160.71 12 in/DG
CL	28.33 12 in/DG	247.0 12 in/QS	213.1 12 in/DG	204.7 5 in/QS	132.7 12 in/DG	15.51 12 in/DG	N/A	N/A	2.38 5 in/QS	139.3 12 in/DG

DB - Design basis limiting result, sequence, and single active failure (SAF).
DEHL - Double ended hot leg break.
DEPS - Double ended pump suction break.
HL - Bounding hot leg break result for 1-12 inch diameter break size spectrum.
CL - Bounding cold leg break result for 1-12 inch diameter break size spectrum.

Attributes:
PP - Peak containment pressure.
PT - Peak containment gas temperature.
MLT - Maximum containment liner temperature.
MST - Maximum sump water temperature: RS for recirculation spray piping, ECCS for ECCS piping.
NPSH - Available net positive suction head: IRS inside recirculation spray (BVPS-1), RS recirculation spray (BVPS-2), ORS outside recirculation spray (BVPS-1), and LHSI low pressure safety injection (BVPS-1).
MSI - Maximum sump inventory.
MSW - Maximum service water heat exchanger outlet temperature.

SAF:
DG - One train each safety injection (SI), quench spray (QS), recirculation spray (RS).
QS - One train QS.
CIB - One train QS and RS.
NF - No failure.
MSCV - Main steam isolation valve.
MFIV - Main feedwater isolation valve.

(1) The designator provides the break description (size or location) preceding the "slash" and SAF following it.

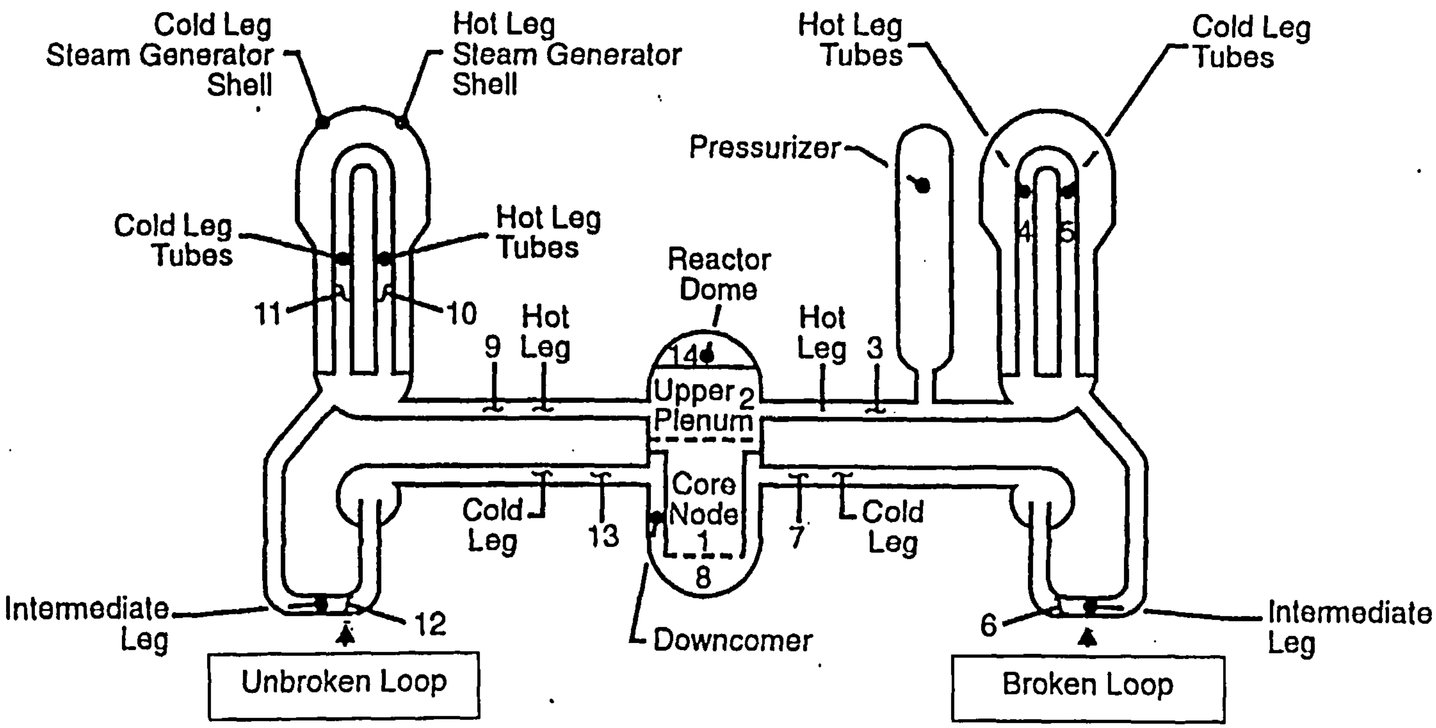


Figure 1: PWR primary system nodalization for Westinghouse design.

Figure 2: Comparison of downcomer water level histories for 2 and 6 inch diameter breaks.

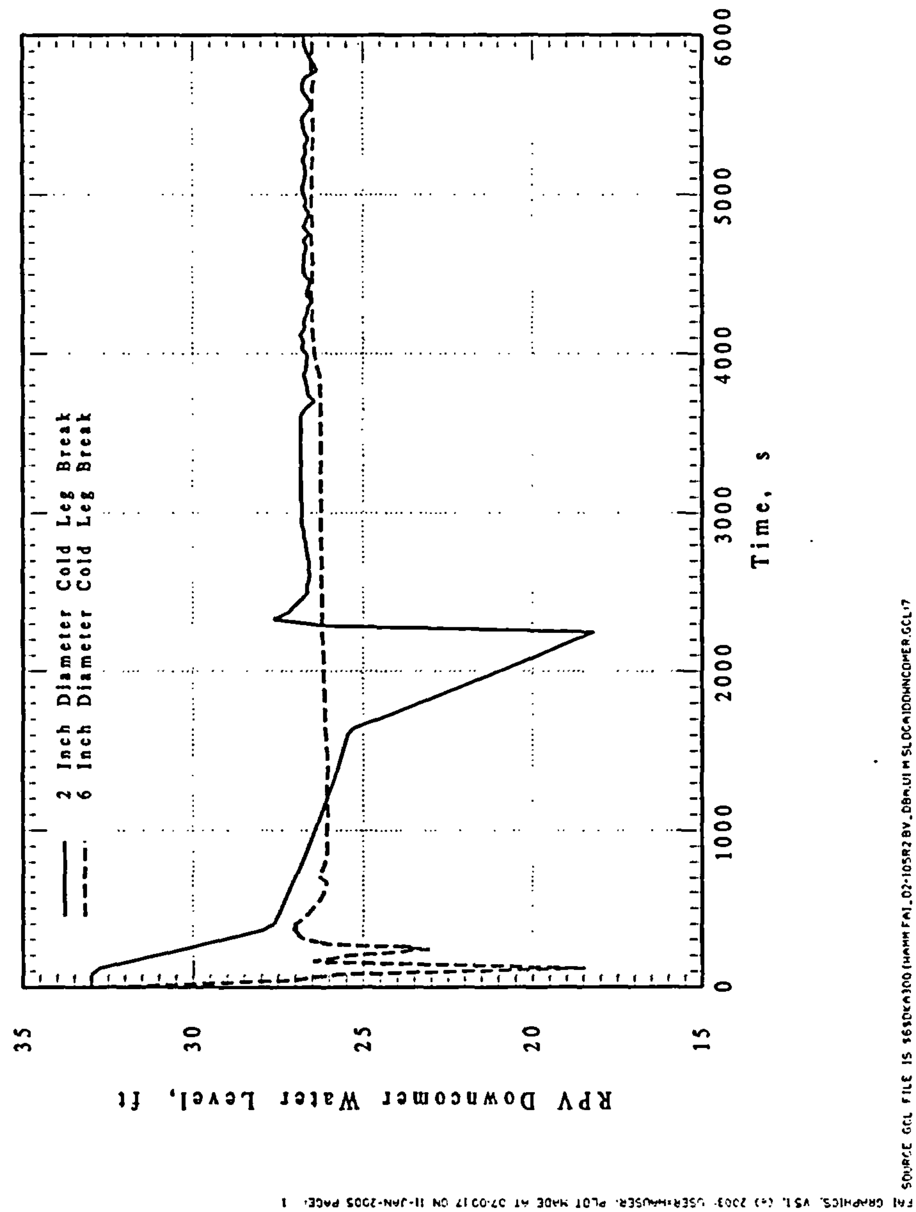


Figure 3: Comparison of predicted vessel discharge mass flux with experiment for 12 inch diameter break.

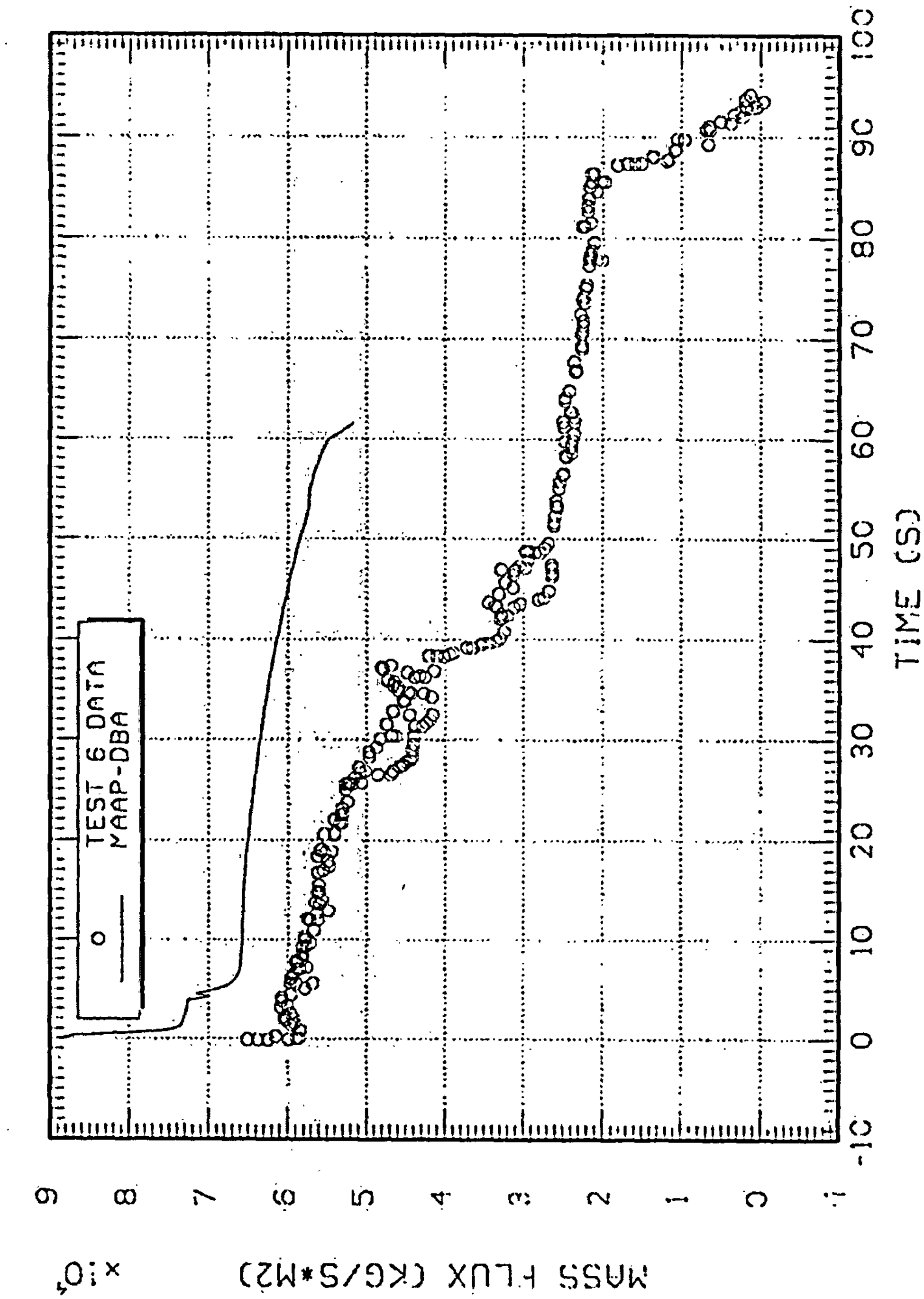
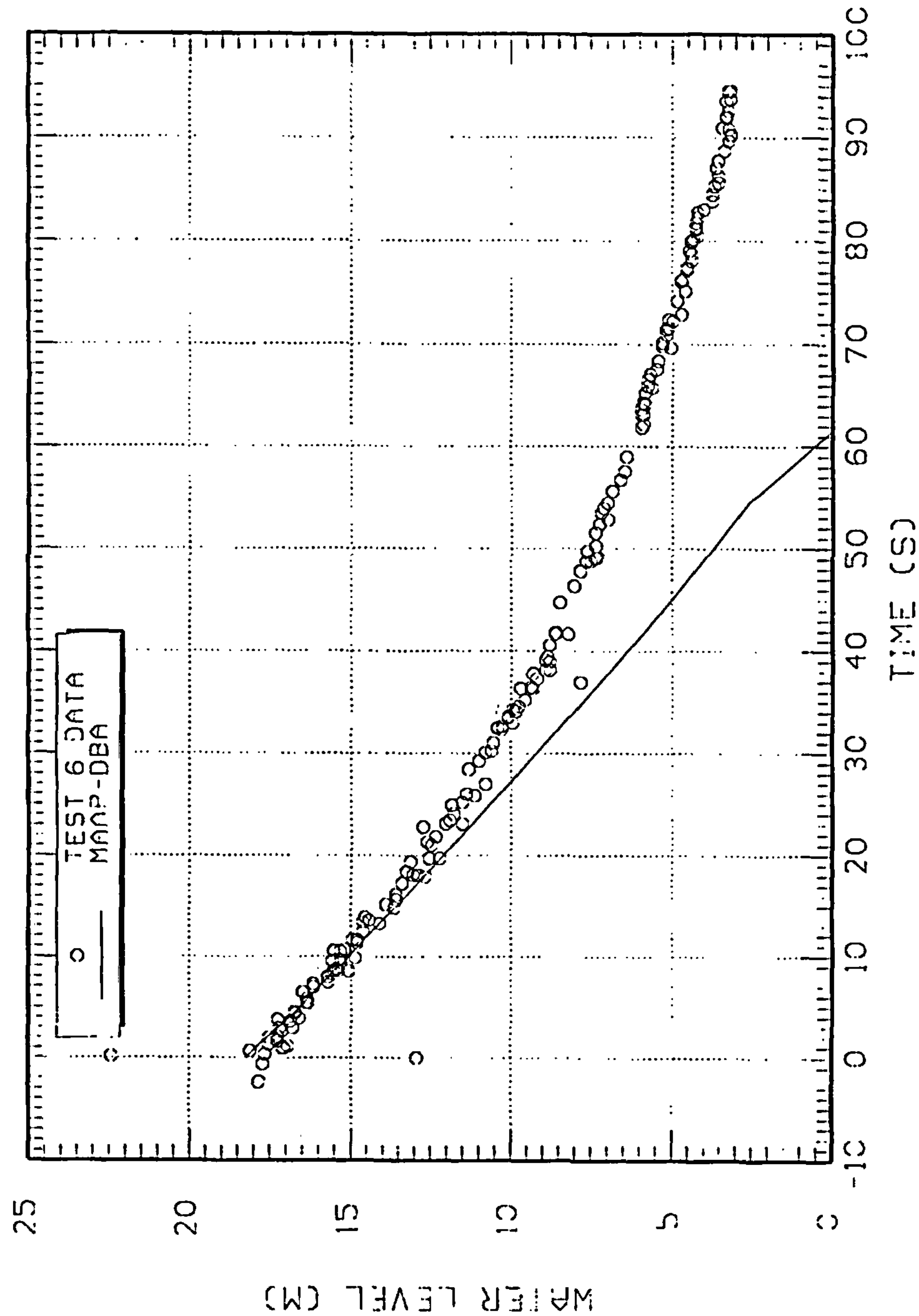


Figure 4: Comparison of predicted vessel water level with experiment for 12 inch diameter break.



21. The minimum containment pressure for the 10 CFR 50.46 LOCA calculations is not discussed in the June 2, 2004, letter. Specify whether this type of calculation is to be covered by the MAAP-DBA review. If so, please describe this calculation.

Response:

The minimum containment pressure calculations for 10CFR 50.46 LOCA analyses will continue to use the Westinghouse COCO program consistent with the Westinghouse methodology.

22. Subcompartment calculation methods are not discussed in the June 2, 2004, letter. Specify whether this type of calculation is to be covered by the MAAP-DBA review. If so, please describe this calculation.

Response:

FirstEnergy Nuclear Operating Company (FENOC) does not intend to perform subcompartment pressure and temperature calculations using the MAAP-DBA methodology. As such, this type of calculation need not be considered as part of the MAAP-DBA review.

23. Enclosure 2 Section 9.2: Describe or provide a reference for the generalized containment model.

Response:

MAAP-DBA (like MAAP4) has the capability to model the containment as a single node or as multiple nodes interconnected by flow junctions. This capability has been referred to as the generalized containment model (GCM) that replaced the fixed node and junctions scheme included in previous versions of MAAP. The rates-of-exchange of mass and energy between multiple nodes are quantified as are the rates-of-exchange of mass and energy with active containment heat removal spray systems as well as passive heat sinks. Moreover, mass and energy balances are performed for each containment node.

A description for the GCM used in MAAP-DBA can be found in the MAAP4 User's Manual, for example consult the descriptions of subroutines AUXREG and AUXFLO. As part of the MAAP-DBA methodology, the treatment of passive containment heat sinks in the GCM was upgraded and is described in Attachment B. An important part of this was a separate evaluation for the steel liners in containment and their paint surfaces where appropriate.

24. Describe the BVPS-1 and -2 quality assurance program to ensure that the MAAP- DBA configuration will be adequately controlled in compliance with 10 CFR Part 50 Appendix B.

Response:

BVPS –1 and 2 implement the requirements of 10CFR Part 50 Appendix B through the Quality Assurance Program Manual (QAPM). The QAPM applies to all activities associated with safety related structures, systems, and components. The QAPM is implemented through the use of approved procedures which provide written guidance for the control of quality related activities and provide for the development of documentation to provide objective evidence of compliance. Both configuration control and model security are controlled.

Computer software procurement, acquisition, upgrade, development, modification, reclassification, relocation, and retirement are controlled under BVPS procedure NOP-SS-1001 "FENOC Administrative Program for Computer Related Activities". That procedure applies to both BVPS-1 and BVPS-2.

NOP-SS-1001 requires that category B software (such as MAPP-DBA) intended for the design or design analysis of safety related structures, systems, or components meet the following requirements:

1. Documentation requirements including:
a. Software Requirements/Functional Specification
b. Software Change Description Requirements
c. Software Design Description
d. Software Testing Requirements
e. Instructions (Users Manual) must be provided
2. Venders must be on the FENOC Approved Suppliers List and 10CFR21 must be applied.
3. Review and Verification in accordance with the Venders Software Assurance Program
4. Testing and Verification must be performed
5. 10 CFR50.59 Process Evaluation
6. Compliance with FENOC Configuration Control Process
7. Software Error Discovery, Notification, and Corrective Action Programs must be applied
8. User Training is required

25. It is proposed to delete Surveillance Requirements (SRs) 4.6.3.1.b and 4.6.3.1.e. Deleting these SRs removes the requirement to cycle each spring- or weight-loaded containment isolation valve and verify that the valve remains closed with < 1.2 psid differential pressure across the valve and opens when the differential pressure is > 1.2 psid but < 6.0 psid. The proposed containment pressure for both units is > 12.8 psia and <14.2 psia which is still slightly subatmospheric.

If the pressure following a DBA returns to this pressure range, show that the function of the check valves is still no longer required.

Response:

The function of the check valves is to prevent in-leakage to the containment when the pressure is sub-atmospheric. Since credit is no longer being taken for sub-atmospheric conditions following a LOCA in order to terminate leakage for the purposes of minimizing offsite dose consequences, it is no longer necessary for the check valves to provide this function.

26. Describe or reference the modeling of the energy exchange during a LOCA between the water on the containment floor (and in the sump) and the containment atmosphere.

Response:

A description of the MAAP-DBA modeling of the energy exchange during a LOCA between the water on the containment floor (and in the sump) and the containment atmosphere can be found in the MAAP User's Manual subroutine HTGPL.

27. In considering the effect your proposed change in containment pressure has on your LOCA analyses, you state that "the evaluation concludes that the LOCA analyses results are not adversely affected" (page 6-2 of your submittal). What is your calculated change due to the proposed containment pressure on fuel peak cladding temperature (PCT) for the most limiting LBLOCA and SBLOCA on BVPS-1 and -2? What has been the deviation in PCT as a result of all accumulated changes to your analyses since the last time your LOCA analyses was reviewed and approved by NRC? The amount of deviation is calculated by summing the absolute value of each of the individual changes. A change of 50°F in any or all accumulated changes since the last time your LOCA analyses were reviewed and approved by Nuclear Regulatory Commission (NRC) is considered significant per 10CFR50.46(a)(3)(i). If the accumulation of all changes to your LOCA analyses (including the increase in containment pressure) shows that your calculated PCT does not differ by more than 50°F, provide information justifying this conclusion. If the accumulation of all changes to your LOCA analyses (including the increase in containment pressure) does differ by more than 50°F, provide details demonstrating compliance with 10CFR50.46 for the most limiting SBLOCA and LBLOCA for both BVPS-1 and -2 including your schedule for submitting a re-analysis if necessary.

Response:

The increased containment pressure associated with containment conversion will result in a reduction in the calculated PCT during a LBLOCA. This is due to an increase in the core flooding rate due to the increased steam venting. It is estimated that the reduction in PCT is 91°F for both units.

The change in containment pressure has no impact on the PCT calculated for SBLOCA. The evaluation methodology is not sensitive to changes in containment backpressure. SBLOCA is analyzed for BVPS using the Westinghouse NOTRUMP evaluation methodology, which does not include containment backpressure modeling. This is not required since the break flow is normally at critical conditions throughout most of the accident and containment pressure therefore does not affect either the break flow or RCS pressure conditions.

The most recent BVPS-1 LBLOCA analysis that was submitted to the NRC was in 1993. This analysis was performed to address changes in allowable steam generator tube plugging levels and was submitted at the request of the NRC although the evaluation was implemented under 10CFR50.59. Since then, a re-analysis was completed in 2002. This analysis was not submitted to the NRC for review and approval; however, completion of this re-analysis was reported in the BVPS 50.46 annual report. The absolute value of all accumulated PCT changes since the 1993 analysis exceeds 50°F. The absolute value of all accumulated changes since the most recent re-analysis in 2002 is 100°F. As reported in our 10CFR50.46 annual report submitted on 11/19/2004, the PCT for BVPS-1 LBLOCA is currently 1996°F. FENOC proposes that a re-analysis will be completed and submitted for BVPS-1 LBLOCA within one year following implementation of containment conversion. It should be noted that an analysis using BELOCA methodology has been submitted to the NRC for BVPS-1, which includes the effects of containment conversion.

The most recent BVPS-2 LBLOCA analysis that was submitted and approved by the NRC was in 1987. Since then re-analyses were completed in 1993 and 2002. These analyses were not submitted to the NRC for review and approval; however, completion of these re-analyses was reported in the BVPS 50.46 annual report. The absolute value of all accumulated PCT changes since the 1987 analysis exceeds 50°F. The absolute value of all accumulated changes since the most recent re-analysis in 2002 is 0°F. As reported in our 10CFR50.46 annual report submitted on 11/19/2004, the PCT for BVPS-2 LBLOCA is currently 2044°F. FENOC proposes that a re-analysis will be completed and submitted for BVPS-2 LBLOCA within one year following implementation of containment conversion. It should be noted that an analysis using BELOCA methodology has been submitted to the NRC for BVPS-2, which includes the effects of containment conversion.

The most recent BVPS-1 SBLOCA analysis that was submitted to the NRC was in 1993. This analysis was performed to address changes in allowable steam generator tube plugging levels and was submitted at the request of the NRC although the evaluation was implemented under 10CFR50.59. Since then, a reanalysis was completed in 2003. This analyses was not submitted to the NRC for review and approval, however, completion of this re-analysis was reported in the BVPS 50.46 annual report. The absolute value of all accumulated PCT changes since the 1993 analysis exceeds 50°F. The absolute value of all accumulated changes since the most recent re-analysis in 2003 is 0°F. As reported in our 10CFR50.46 annual report submitted on 11/19/2004, the PCT for BVPS-1 SBLOCA is currently 1849°F.

The most recent BVPS-2 SBLOCA analysis that was submitted and approved by the NRC was in 1987. Since then, re-analyses were completed in 1993 and 2003. These analyses were not submitted to the NRC for review and approval; however, completion of these re-analyses was reported in the BVPS 50.46 annual report. The absolute value of all accumulated PCT changes since the 1987 analysis exceeds 50°F. The absolute value of all accumulated changes since the most recent re-analysis in 2003 is 0°F. As reported in our 10CFR50.46 annual report submitted on 11/19/2004, the PCT for BVPS-2 SBLOCA is currently 2105°F.

28. For both BVPS-1 and -2, provide details from your analyses justifying your conclusion that for the feedline break event the proposed containment pressure would either have no impact or provide a very slight benefit. Auxiliary feedwater (AFW) is used to mitigate this event and is initiated on low-low steam generator level. Provide details that justify whether or not AFW will be affected as a result of the proposed change in containment pressure. If it is affected, describe in detail how this does not cause adverse consequences to the mitigation of this event. For BVPS-1, provide details from your analyses (in addition to what is in your submittal on page 6-7) on how the installation of the cavitating venturis in conjunction with the proposed containment pressure would not adversely affect the mitigation of this event.

Response:

The break release model for feedline break assumes a constant containment backpressure. Generally at higher steam generator pressures, the break flow is independent of the containment pressure since the flow is choked. A slightly higher containment pressure resulting from implementation of containment conversion may have a slight benefit due to a lower release flow, but most likely there would be no impact on the results. The auxiliary feedwater is not affected by the change in containment backpressure. In the current feedline break analysis, all AFW flow is assumed to be lost to the broken feedline until operator action occurs to isolate flow to this path. Following isolation, flow to the intact steam generators commences. The flow to the intact steam generators is calculated based on the safety valve set pressure (including tolerances) and is not affected by containment pressure. The installation of the cavitating venturis to support an atmospheric containment design does not impact this calculated flow to the intact steam generators since adequate margin exists in the AFW pump to accommodate the additional pressure drop (approximately 42 psid) in the system and still provide the currently assumed flow. However, the addition of the venturis limits the flow to the broken feedline prior to operator action to isolate the flow to this path, which affords additional margin since flow to the intact steam generators will occur prior to isolation. This additional margin has not been credited in the existing calculation but is included in the analysis performed to support EPU.

29. You propose changes to the TSs in which the refueling water storage tank level setpoints for switching from safety injection to recirculation mode for BVPS-1 and 2 have decreased. Provide the details from your analyses that demonstrate that you will have adequate NPSH for your low head safety injection and quench spray pumps at the lowest allowed levels (13 feet 9 inches for BVPS-1 and 31 feet 8 inches for BVPS-2). Also provide the details of your analyses that demonstrate that you will not experience vortexing at the lowest allowed levels.

Response:

The minimum setpoint for the low level setpoint on the refueling water storage tank (RWST) is limited by the available NPSH to the HHSI pumps for both Units. Revised NPSH analyses for the LHSI and HHSI pumps were performed to reflect the change in setpoints for both BVPS-1 and -2. The switchover setpoint is not the limiting point for the Quench Spray (QS) pumps since these pumps run until shut off by operator action when the RWST is nearly empty. Therefore, the limiting NPSH is not affected by the changes in the switchover setpoint. Table 4 shows a summary of the NPSH parameters and results based on the revised setpoints for the HHSI, LHSI and QS pumps.

Based on the height of water above the suction nozzles for the HHSI, LHSI and QS pumps at the switchover to recirculation level, vortexing will not occur. The minimum height of water above the suction nozzles for the SI and QS pumps is over 10 feet on BVPS-1 and over 28 feet on BVPS-2. This exceeds the minimum submergence recommended in Regulatory Guide 1.82 Revision 3 Table A-1 for zero air ingestion. Scale model testing on the BVPS-1 QS suction nozzle configuration has shown that these recommendations are very conservative and much lower submergence levels can be experienced without vortexing or resultant air ingestion.

Table 4

Summary of the NPSH parameters and Results

Pump	BVPS-1 HHSI	BVPS-1 LHSI	BVPS-1 QS	BVPS-2 HHSI	BVPS-2 LHSI	BVPS-2 QS
1.RWST Low Low setpoint elevation (switchover)(feet)*	749.24	749.24	749.24	766.64	766.64	766.64
2.Pump Suction Elevation (feet)	725.92	684.0	737.7	738.92	720.5	720.5
3.Static Head Available [1-2] (feet)	23.32	64.24	11.54	27.72	46.14	46.14
4.RWST Pressure (atmospheric) feet (psia)	33.17 (14.36)	33.17 (14.36)	33.17 (14.36)	33.17 (14.36)	33.17 (14.36)	33.17 (14.36)
5.Suction Line Friction Loss at maximum flow(feet)	9.04	14.8	4.0	16.29	10.34	5.1
6.Vapor Pressure at Max RWST Temperature (65 F) feet (psia)	0.704 (0.305)	0.704 (0.305)	0.704 (0.305)	0.704 (0.305)	0.704 (0.305)	0.704 (0.305)
NPSH Available [3+4-5-6] (feet)	46.7	81.9	40.0	43.9	68.3	73.5
NPSH Required at maximum flow (feet)	45	12	25	40	18	38

*Minimum switchover elevation including instrument uncertainty

L-05-006 Attachment B

MAAP-DBA
One-Dimensional Heat
Conduction in a Plane Wall

ONE-DIMENSIONAL HEAT CONDUCTION IN A PLANE WALL

1.0 BACKGROUND

Integral assessments of the containment response resulting from the surge of steam or a flashing two-phase mixture into the containment atmosphere involve the thermal conduction of energy into the metallic and concrete heat sinks. In particular, thermal conduction into the containment liner and subsequently into the concrete behind the liner is an important element of how the containment would respond to these DBA conditions and is a specific assessment for the maximum liner temperature. Each heat sink in the MAAP model can be represented as either one or two-sided structure and one or both sides could be covered by a painted steel liner. Of course, there are also stainless steel liners such as that for the refueling pool which are not painted; these are represented as unpainted stainless steel. MAAP-DBA evaluates the conduction into metal and concrete heat sinks through the transient one-dimensional model discussed below.

2.0 GENERAL DESCRIPTION

This dynamic benchmark compares the numerical solution used in the MAAP-DBA model to the analytical solutions for one-dimensional transient conduction. In particular, the numerical values for conduction through a steel slab as well as into a 0.3 m thick concrete are compared with the analytical solution.

3.0 PHYSICAL BASIS OF THE MODEL

MAAP-DBA models heat transfer from a gas or water region to structures (such as, steel walls, concrete walls, or concrete walls with steel liners) in containment and conduction within the structures. Once the convective and radiation heat transfer from the surrounding gas or water to the heat sink surface is calculated, the conduction within the structure is calculated using an implicit one-dimensional finite difference scheme.

For a slab, the one-dimensional conduction equation without any volumetric heat generation is written as

$$\rho c \frac{\partial T}{\partial t} = \frac{\partial}{\partial x}\left(k \frac{\partial T}{\partial x}\right) \qquad \text{for } 0 < x < L \tag{1a}$$

$$-k \frac{dT}{dx} = \left[h_1 \left(T_{\infty,1} - T\right) \text{or } q_1''\right] \qquad \text{at } x = 0 \tag{1b}$$

$$k \frac{dT}{dx} = \left[h_2 \left(T_{\infty,2} - T\right) \text{or } q_2''\right] \qquad \text{at } x = L \tag{1c}$$

where q" (i = 1, 2) is the overall heat flux imposed on the slab walls. Discretizing equation (1), we obtain the finite difference equations for interior nodes and boundary nodes.

3.1 Interior Nodes

The formulation of the implicit finite difference scheme for interior nodes, shown in Figure 1, is written as [Patankar, 1980]

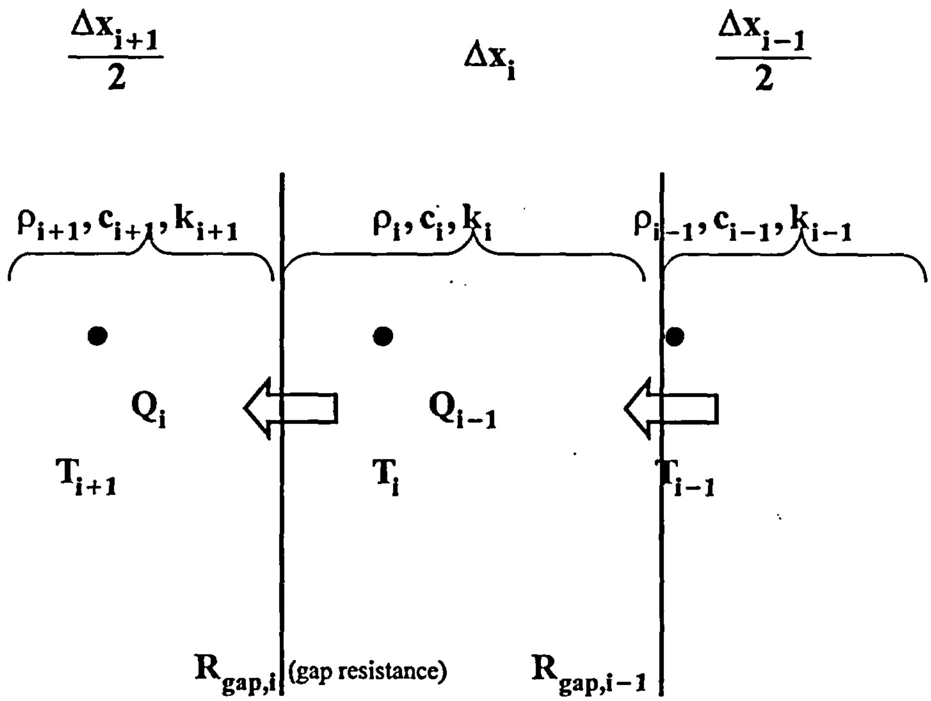
$$\rho_i\, c_i \frac{\Delta x_i}{\Delta t}\left(T_i - T_i^{old}\right) = \frac{T_{i-1} - T_i}{\dfrac{\Delta x_{i-1}}{2\,k_{i-1}} + \dfrac{\Delta x_i}{2\,k_i} + R_{gap,i-1}} - \frac{T_i - T_{i+1}}{\dfrac{\Delta x_i}{2\,k_i} + \dfrac{\Delta x_{i+1}}{2\,k_{i+1}} + R_{gap,i}} \tag{2}$$

where $R_{gap,i}$ is the gap resistance between nodes "i" and "i + 1", T_i^{old} is the temperature of node i at the beginning of time step, and the rest of the temperatures are defined at the end of the time step. Effective thermal resistance, R_i, between the centers of nodes "i-1" and "i" can be defined as

$$R_i = \frac{\Delta x_{i-1}}{2\,k_{i-1}} + \frac{\Delta x_i}{2\,k_i} + R_{gap,i-1} \tag{3}$$

Likewise,

Figure 1, Interior heat sink node.



$$R_{i+1} \quad = \quad \frac{\Delta x_i}{2\,k_i} + \frac{\Delta x_{i+1}}{2\,k_{i+1}} + R_{gap,i} \text{ between "i" and "i+1"} \tag{4}$$

Using the effective thermal resistances, equation (2) is reduced to between "i" and "i+1"

$$\rho_i\, c_i\, \frac{\Delta x_i}{\Delta t}\left(T_i - T_i^{old}\right) \quad = \quad \frac{T_{i-1} - T_i}{R_i} - \frac{T_i - T_{i+1}}{R_{i+1}} \tag{5}$$

The above equation is rearranged to the following form:

$$A_i\, T_{i-1} + B_i\, T_i + C_i\, T_{i+1} \quad = \quad D_i \tag{6}$$

where $A_i \quad = \quad -\dfrac{1}{R_i}\,,$ (6a)

$$B_i \quad = \quad \left(\rho_i\, c_i\, \frac{\Delta x_i}{\Delta t} + \frac{1}{R_i} + \frac{1}{R_{i+1}}\right), \tag{6b}$$

$$C_i \quad = \quad -\frac{1}{R_{i+1}}\text{, and} \tag{6c}$$

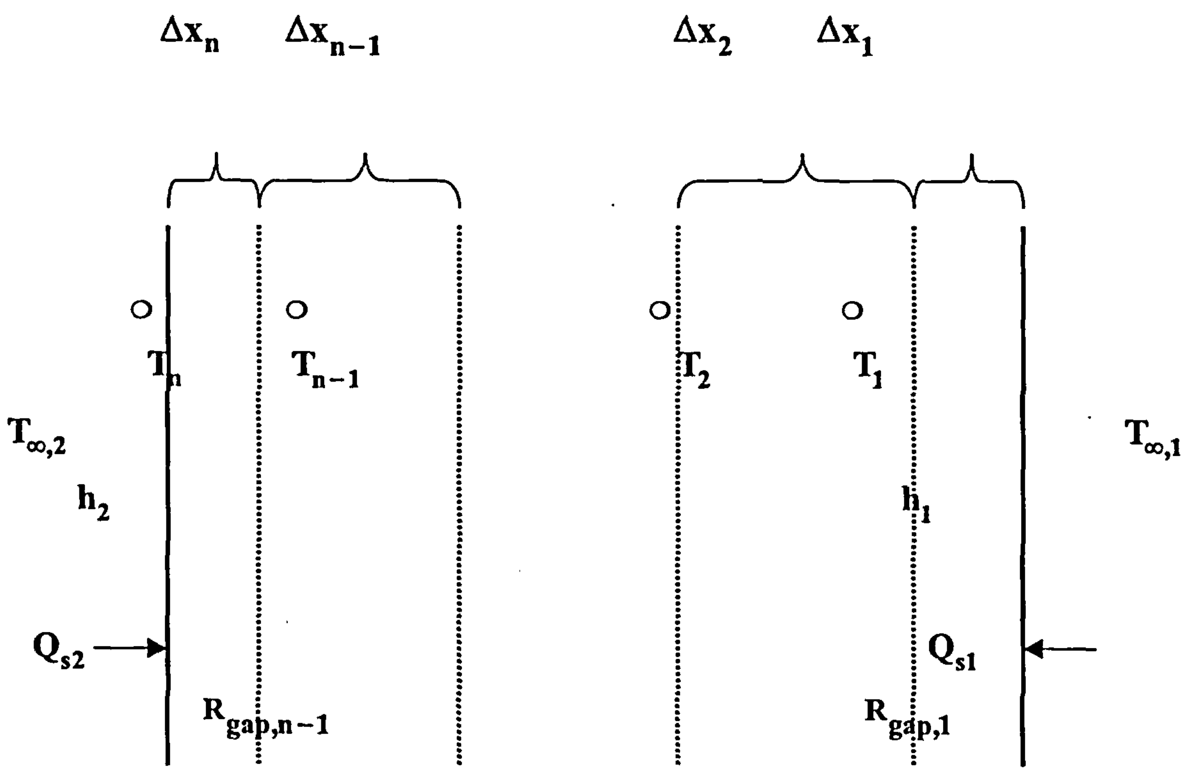
$$D_i \quad = \quad \rho_i\, c_i\, \frac{\Delta x_i}{\Delta t}\, T_i^{old}\,. \tag{6d}$$

3.2 Boundary Nodes

For boundary nodes, "half" nodes are used and temperatures are defined at the surfaces as shown in Figure 2. By replacing conduction path to an appropriate boundary condition, the following equation for node 1 can be written

$$\rho_i\, c_i\, \frac{\Delta x_i}{\Delta t}\left(T_i - T_i^{old}\right) \quad = \quad -\frac{T_1 - T_2}{\dfrac{\Delta x_1}{k_1} + \dfrac{\Delta x_2}{2\,k_2} + R_{gap,1}} + \left[\frac{Q_{s1}}{A_{hs}} \text{ or } h_1\left(T_{\infty,1} - T_1\right)\right] \tag{7}$$

Figure 2, Boundary half-nodes.



where Q_{s1} is the external heat source to the surface 1, h_1 is the heat transfer coefficient, A_{hs} is the heat transfer area, and $T_{\infty,1}$ is the environment temperature facing surface 1. The effective resistance R_1 is defined as

$$R_1 = \frac{\Delta x_1}{k_1} + \frac{\Delta x_2}{2\,k_2} + R_{gap,1} \tag{8}$$

Then, equation (7) for surface 1 becomes

$$\rho_i\, c_i \frac{\Delta x_1}{\Delta t}\left(T_1 - T_1^{old}\right) = -\frac{T_1 - T_2}{R_1} + \left[\frac{Q_{s1}}{A_{hs}} \text{ and/or } h_1\left(T_{\infty,1} - T_1\right)\right] \tag{9}$$

and reduces to

$$A_1\, T_0 + B_1\, T_1 + C_1\, T_2 = D_1 \tag{10}$$

where $A_1 = 0$,

$$B_1 = \rho_1\, c_1 \frac{\Delta x_1}{\Delta t} + \frac{1}{R_1} + h_1\,,$$

$$C_1 = -\frac{1}{R_1}\,, \text{ and}$$

$$D_1 = \rho_1\, c_1 \frac{\Delta x_1}{\Delta t} T_1^{old} + \left[\frac{Q_{s1}}{A_{hs}} \text{ and/or } h_1\, T_{\infty,1}\right].$$

In the current MAAP-DBA containment application, h_1 is set to zero and Q_{s1} is defined as a total heat input to surface 1, which includes heat inputs due to steam condensation, convection, and radiation heat transfer. Similar to node 1, the equation for node "n" is reduced to

$$A_n\, T_{n-1} + B_n\, T_n + C_n\, T_{n+1} = D_n \tag{11}$$

where $R_{n-1} = \dfrac{\Delta x_{n-1}}{2\,k_{n-1}} + \dfrac{\Delta x_n}{k_n} + R_{gap,n-1}\,,$

$$C_n = 0\,,$$

$$A_n = -\frac{1}{R_{n-1}}\,,$$

$$B_n = \rho_n c_n \frac{\Delta x_n}{\Delta t} + \frac{1}{R_{n-1}} + h_2 ,$$

$$D_n = \rho_n c_n \frac{\Delta x_n}{\Delta t} T_n^{old} + \left[\frac{Q_{s2}}{A_{hs}} \text{ and/or } h_2 T_{\infty,2} \right], \text{and}$$

h_2 = heat transfer coefficient on surface 2.

Similar to surface 1, h_2 is set to zero and Q_{s2} is defined as a total heat input to surface 2.

The solution of these sets of linear equations for the end-of-time step temperatures is obtained by the Tri-Diagonal Matrix Algorithm.

3.3 Individual Mesh Size

In MAAP-DBA, the maximum number of nodes for a given wall is increased to 40, compared to 20 used in MAAP4. Depending on the material and the total thickness, the total number of nodes for a given wall can be less than 40. For individual node size, a different scheme is used for steel than for concrete. For steel, uniform node size is used, except for "half" nodes at the boundaries. The size of the steel node is given as

$$\Delta x_{steel} = \max\left(0.0025\,\text{m}, \sqrt{2\,\alpha_{steel}\, t_{ref}}\right) \tag{12}$$

where t_{ref} is set to 1 second. The above node size represents the minimum node thickness with the time step of 1 second when the explicit solution scheme is employed. Even though the implicit solution scheme is used here, the stability criterion for an explicit solution scheme with 1 second time step is adopted as a simple criterion to determine the node thickness. When the total thickness is less than Δx_{steel}, the steel wall is modeled as one node. The same approach is used to determine the liner node thickness $\Delta x_{w,s}$. The maximum number of nodes for a given liner is limited to 5.

When a concrete wall is subjected to a high heat flux due to radiation from the molten debris pool or during the initial blowdown phase of a design basis accident, a steep temperature gradient results near the wall surface. To model the possible steep temperature gradient, the nodalization scheme for concrete includes ten fine nodes near each face of a two-sided wall and near face 1 of a one-sided wall. The number of coarse nodes is limited to 40, minus the number of fine nodes and the number of liner nodes, if any.

The size of the fine concrete node, $\Delta x_{w,f}$, is given as

$$\Delta x_{w,f} = \max\left(0.0025\,\text{m}, 4\sqrt{\alpha_{concrete}\, t_{ref}}\right) \tag{13}$$

The above concrete node size represents the approximate thermal penetration length for 1 second. For example, the fine concrete node size with $\alpha_{concrete}$ = 6.875E-7 m^2/s becomes about 3.3 mm. The remaining thickness (overall thickness – total thickness of fine concrete node) divided by the number of coarse nodes determines the size of the coarse concrete, $\Delta x_{w,c}$. Figures 3 and 4 show examples of the wall nodalization scheme for a two-sided concrete wall.

4.0 VALIDATION

The 1-D conduction model was validated solving for the transient temperature profile in a one-dimensional slab exposed at time zero to a fluid at constant temperature T_∞ on one surface with a constant heat transfer coefficient h_∞, and insulated on the other surface. Simulation of this problem uses a vertical one-sided wall in the environment node initially at 300 K submerged under water at T_∞ with an externally provided heat transfer coefficient h_∞. The MAAP calculation is compared against analytical solutions that used a constant heat transfer coefficient.

Figure 3, Nodalization scheme for a two-sided concrete wall without liners.

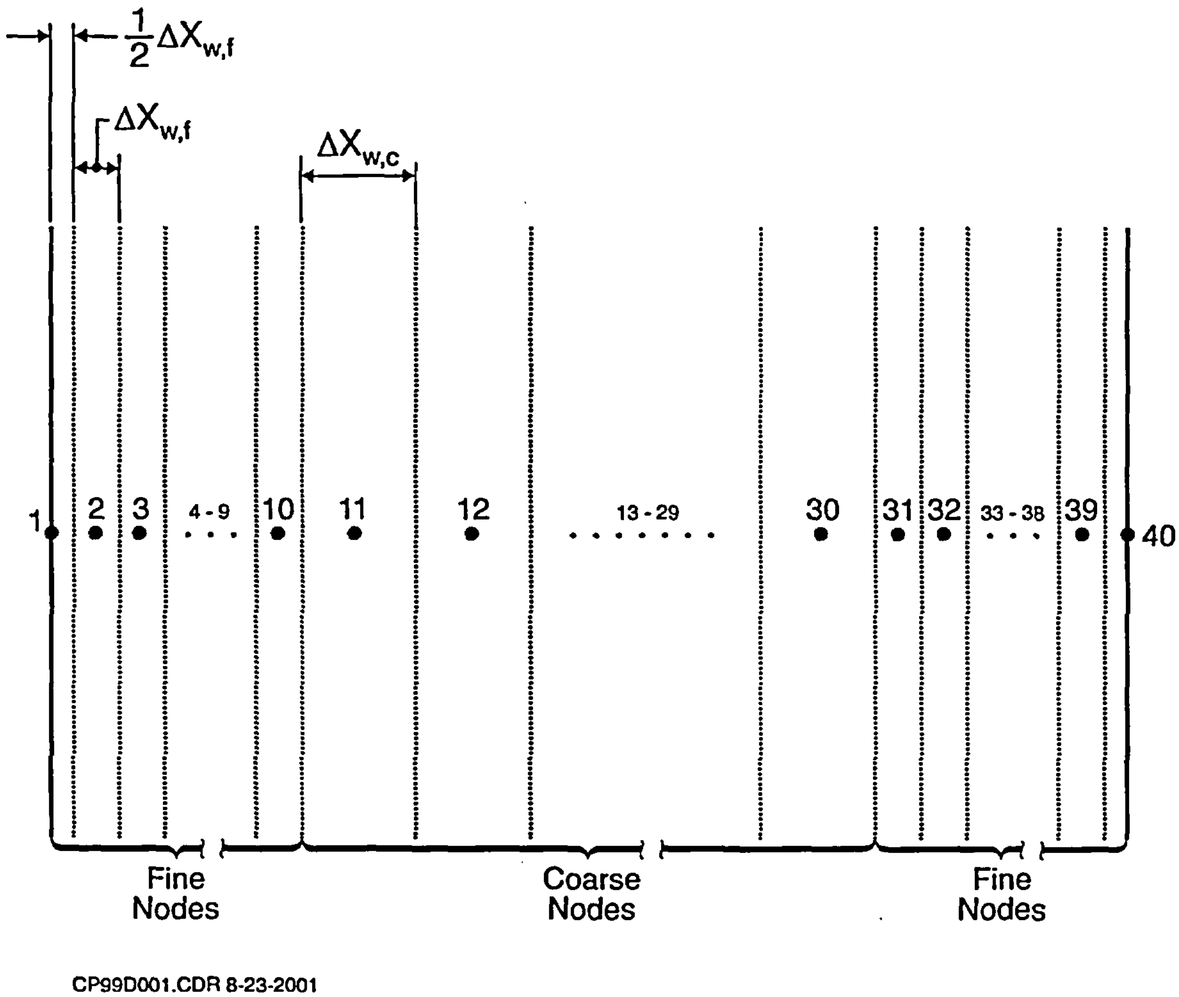
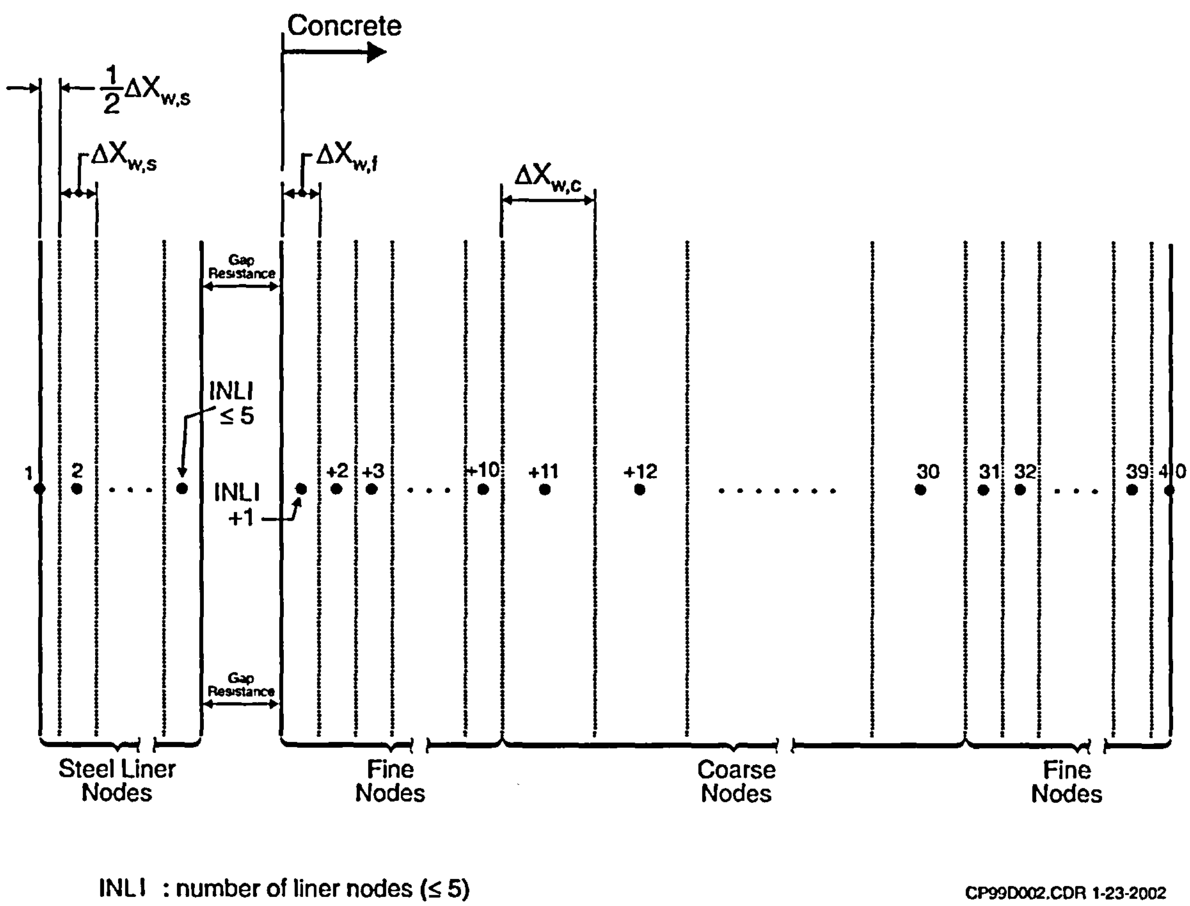


Figure 4, Nodalization scheme for a two-sided concrete wall with a liner on the left side.



The slab has thickness L, uniform initial temperature T_0, environment temperature T_∞, thermal conductivity k, specific heat c, and density ρ. Two different cases were run to validate the MAAP model:

Case 1	Case 2
L = 10 cm steel slab	L = 0.3 m concrete
T_0 = 300 K	T_0 = 300 K
T_∞ = 380 K	T_∞ = 380 K
h_∞ = 200 W/m^2-K	h_∞ = 200 W/m^2-K
k = 45 W/m-K	k = 1.3846 W/m-K
c = 502 J/kg-K	c = 879 J/kg-K
ρ = 7849 kg/m^3	ρ = 2290.64 kg/m^3

Figures 5 and 6 show comparisons of the temperature profiles calculated by MAAP and the analytical solutions. Results are in good agreement.

5.0 NOMENCLATURE

Nomenclature is defined at first appearance in text.

6.0 REFERENCES

Patankar, S. V., 1980, "Numerical Heat Transfer in Fluid Flow," Hemisphere Publishing, New York.

Figure 5, Comparison of the temperature profile calculated by MAAP and analytical solution for a 0.1 m steel.

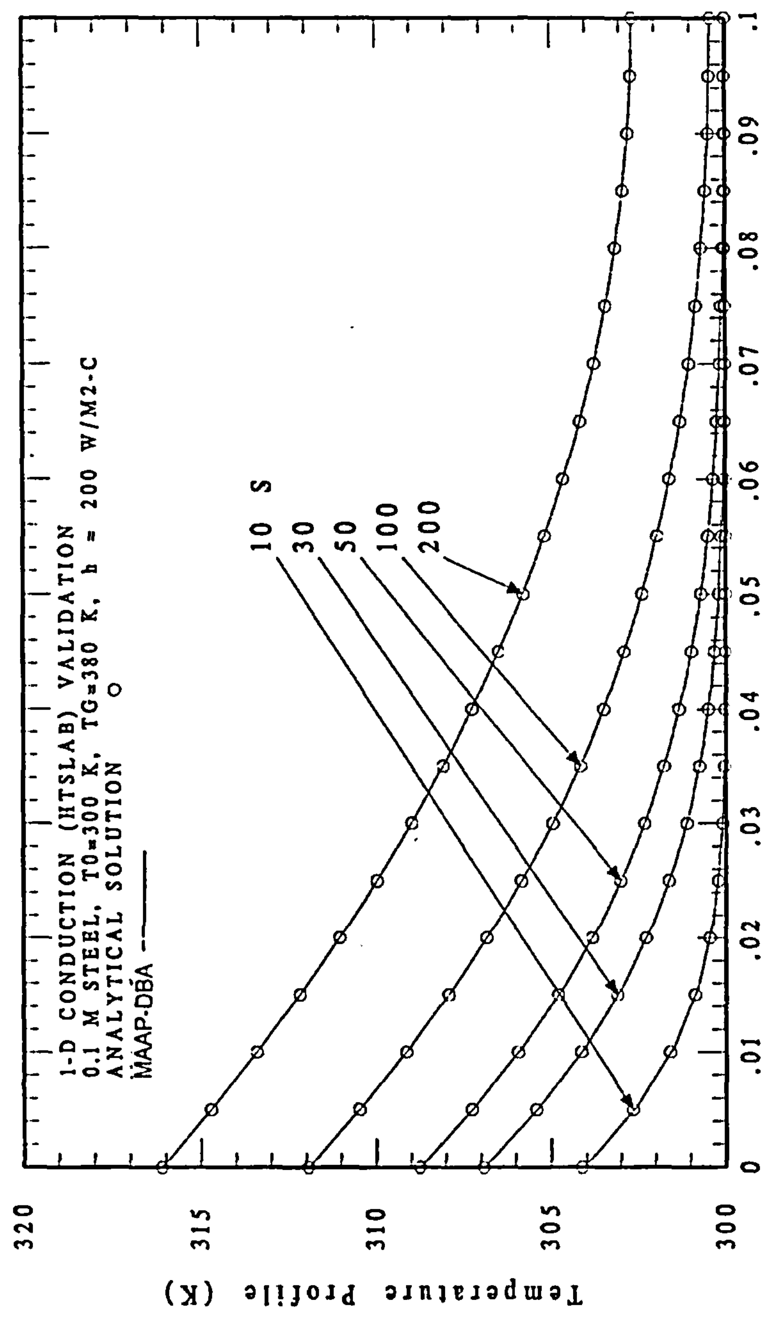
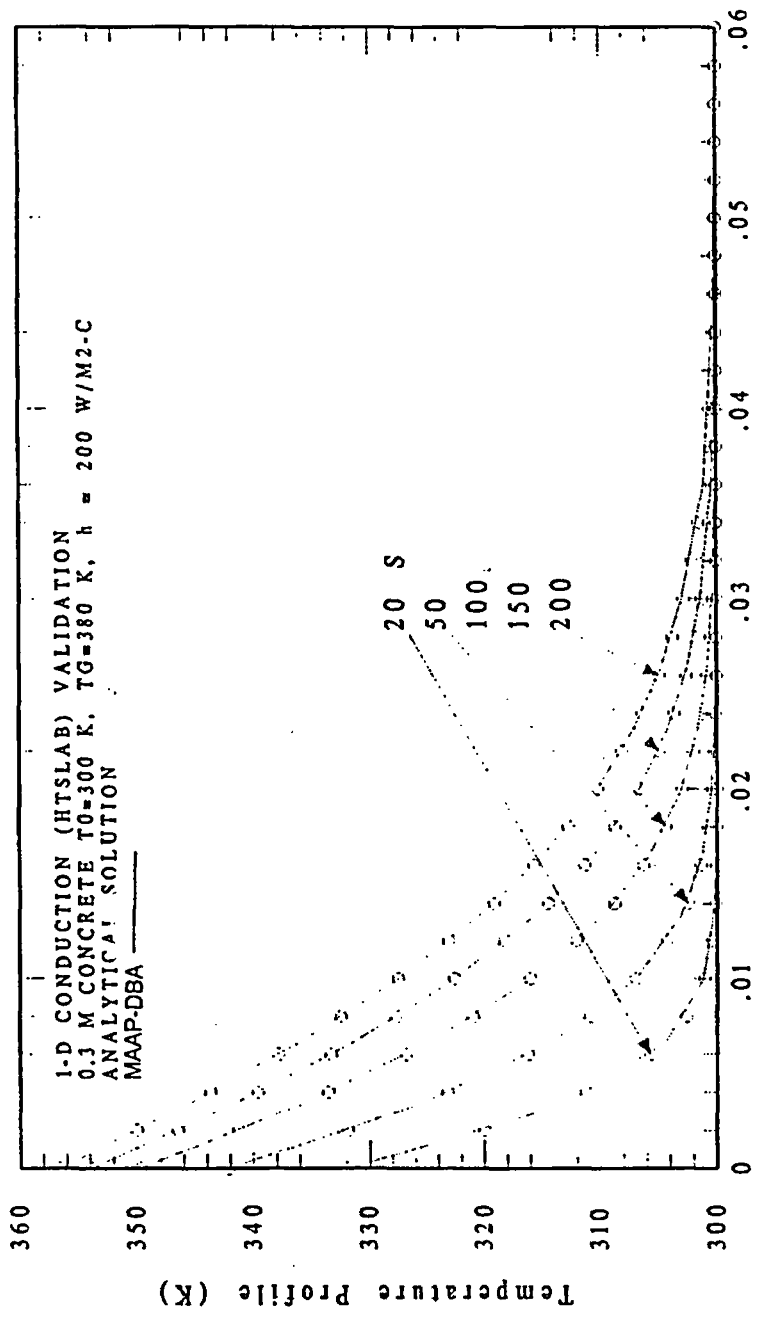


Figure 6, Comparison of the temperature profile calculated by MAAP and analytical solution for a 0.3 m concrete.



L-05 -006 ATTACHMENT C

Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC. They are described only as information and are not regulatory commitments. Please notify Mr. Henry L. Hegrat, Supervisor - Licensing at 330-315-6944 of any questions regarding this document or associated regulatory commitments.

Commitment	Due Date
Complete and submit a re-analysis of Large Break LOCA for BVPS Unit No. 1.	Within one year following implementation of Unit No. 1 containment conversion.
Complete and submit a re-analysis of Large Break LOCA for BVPS Unit No. 2.	Within one year following implementation of Unit No. 2 containment conversion.