FENOC

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January 25, 2006
L-06-003

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

Subject: **Beaver Valley Power Station, Unit Nos. 1 and 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Additional Information in Support of License Amendment Request Nos. 302 and 173 (Unit No. 1 TAC No. MC4645/Unit No. 2 TAC No. MC4646)

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

Enclosures 1 through 3 provide additional information that pertains to the EPU LAR. The additional information is the result of various actions associated with the review of the EPU LAR. The reason for the additional information is provided in each of the enclosures.

Enclosure 1 contains the FENOC responses to the additional questions relative to containment overpressure credit and the risk impact as a result of EPU, provides clarification of the results presented in response to Question 2.c of FENOC letter L-05-192 (Reference 2), and provides the additional information regarding MAAP analysis for Station Blackout (SBO).

Enclosure 2 contains the FENOC response to the additional question relative to operator action times previously identified in FENOC letter L-05-154 (Reference 3).

Enclosure 3 contains additional information requested as a follow-up to the November 29-30, 2005, EPU Dose Assessment Audit regarding Control Room atmospheric dispersion factors and dose consequence information provided in the EPU LAR (Reference 1) as amended by FENOC Letter No. L-05-204 (Reference 4).

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The responses and additional information provided by this transmittal have no impact on either the proposed Technical Specification changes or the no significant hazards consideration transmitted by Reference 1.

No new regulatory commitments are contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager – FENOC Fleet Licensing, at (330) 315-7243.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 25, 2006.

Sincerely,

James H. Lash

Enclosures:

1. PRA – Human Reliability and Containment Overpressure
2. Operator Action Times
3. Additional Information Regarding the Control Room Atmospheric Dispersion Factors and Dose Consequence Information

References:

1. FENOC Letter L-04-125, License Amendment Request Nos. 302 and 173, dated October 4, 2004.
2. FENOC Letter L-05-192, Supplemental PRA Information in Support of License Amendment Request Nos. 302 and 173, Extended Power Uprate (EPU), dated December 9, 2005.
3. FENOC letter L-05-154, Supplemental Information for License Amendment Request Nos. 302 and 173, dated October 7, 2005.
4. FENOC letter L-05-204, Additional Information Regarding Dose Consequence Analyses in Support of License Amendment Request Nos. 302 and 173, dated December 30, 2005.

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

Enclosure 1 of L-06-003

PRA – Human Reliability and Containment Overpressure

The NRC reviewers requested additional information regarding the FENOC submittal (L-05-192) dated September 9, 2005, Supplemental PRA Information in Support of License Amendment Request Nos. 302 and 173 Extended Power Uprate (EPU).

Reason for the contained additional information:

During a telephone call held on January 9, 2006, with the NRC reviewers, additional information and clarification was requested regarding the request for additional information (RAI) response 2.c submitted previously by FENOC letter L-05-192 (Reference 1)

This enclosure contains the FENOC responses to the additional questions relative to containment overpressure credit and the risk impact as a result of EPU, provides clarification of the results presented in response to Question 2.c of Reference 1, and provides the additional information regarding MAAP analysis for Station Blackout (SBO). Specifically, the following questions were asked:

Note:
Responses to Questions 1 and 2 below will be provided in a later supplement.

Question:

1. Justify that the following operator actions can be completed within the time frame from receipt of the cue for the action to the point at which an irreversible plant state leading to core damage is reached under EPU conditions. List the key action steps for each action. Describe whether the actions take place in the control room or outside the control room. Provide the basis for the conclusion that the time available is sufficient to complete the action (e.g., information from simulator observations, job performance measures, walk-through, talk-through, etc.).
(Important operator actions with relatively short time available were identified by the NRC and provided to the licensee.)

Question:

2. Please provide additional information as detailed below:

Question:

2.a. Unit 1 OPRCD4 - No modular accident assessment program (MAAP) analysis is referenced for this operator action. What is the basis for the reduction in human error probability from 8.3E-2 in BV1REV3 to 5.1E-2 in the "EPU RAI" model?

2.b. Unit 1 OPRMU2 and OPRMU5 - Why is the time available for refueling water storage tank make-up much shorter (0.79 hours) for small break loss of coolant accident (LOCA) than the time available (7 hours) for an inter-system LOCA? Explain why the small break LOCA human error probability for this action (6.25E-3) is smaller than for the inter-system LOCA (1.01E-2).

2.c. Unit 1 OPROS1 and OPROS2 - Why is there less time available (0.72 hours) for manually actuating safety equipment during a transient event (OPROS1) than the time

available (0.94 hours) for the same action, given a small LOCA or steam line break (OPROS2)?

2.d. Unit 2 OPRMU2 - Why is the pre-EPU time available (1.55 hours) less than the post-EPU time available (2.58 hours), given that both times were determined using MAAP?

2.e. Unit 2 OPROS2 - Why is the pre-EPU time available (0.89 hours) less than the post-EPU time available (0.94 hours), given that both times were determined using MAAP?

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Credit for Containment Pressure in Emergency Core Cooling Systems (ECCS) and Containment Heat Removal Pumps Net Positive Suction Head (NPSH) Analysis:

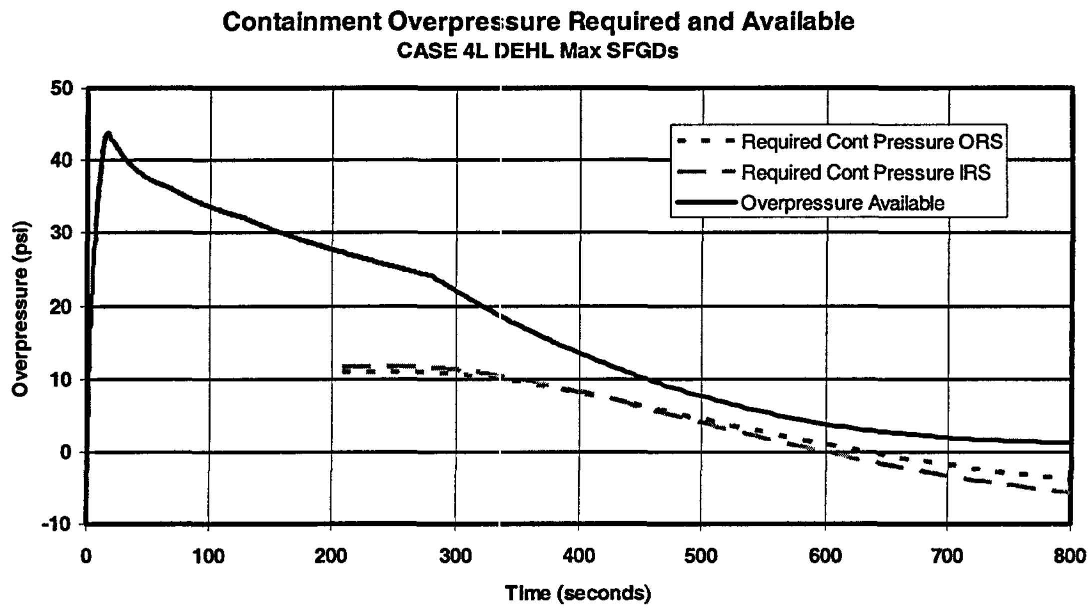
Question:

3. What is the current and the requested values of available NPSH for all ECCS and containment heat removal pumps? Provide these values as a function of time and show the amount of time that containment accident pressure is credited for available NPSH. Compare these available NPSH values with the pre-EPU values for both magnitude and duration.

Response:

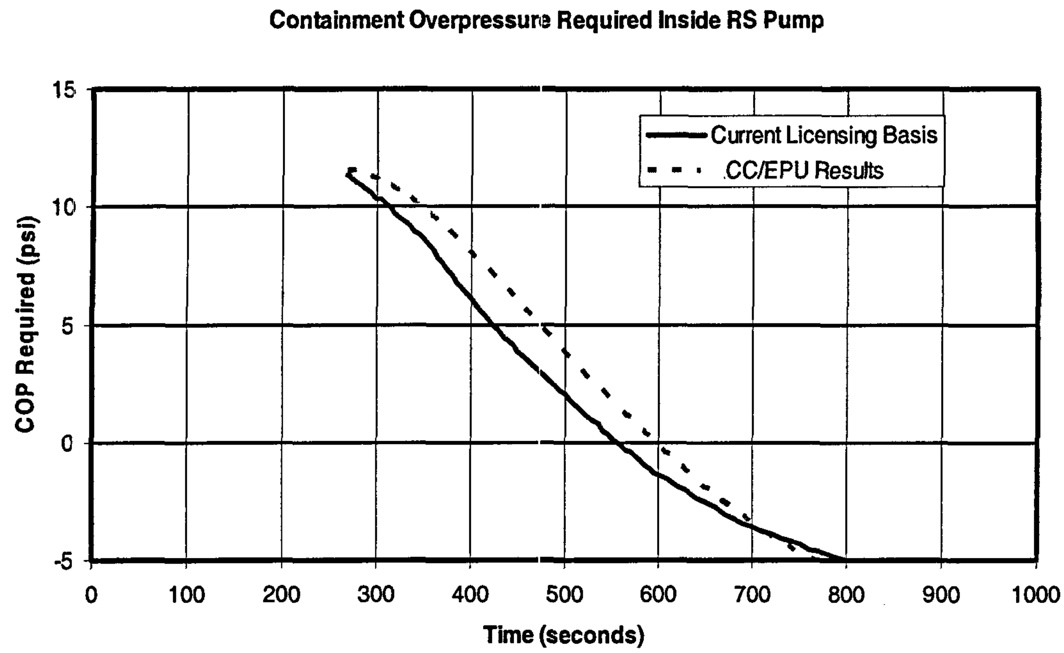
The following plot shows the Containment Overpressure (COP) required for the Inside Recirculation Spray (IRS) pumps and the Outside Recirculation Spray (ORS) pumps, and the available overpressure for this case. The function of the Recirculation Spray (RS) system at Beaver Valley Power Station Unit No. 1 (BVPS-1) is to assist the Quench Spray system in reducing containment pressure following a LOCA and provide containment heat removal. The system consists of two IRS pumps and two ORS pumps which take suction from the containment sump and deliver flow to containment spray headers. The pumps start on a fixed time delay following a high containment pressure signal. Each flow path contains a heat exchanger cooled by service water which provides the only means of containment heat removal for long term cooling of the containment sump water.

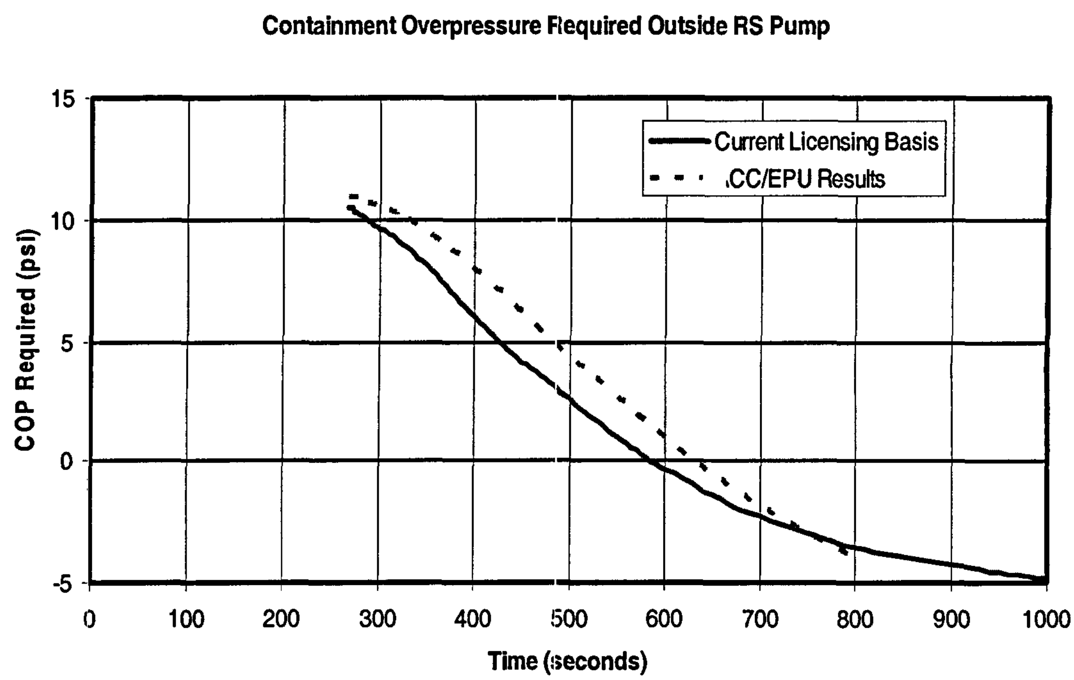
The term "COP Required" has been calculated as that amount of pressure above the initial pressure in containment at the start of the accident necessary to ensure that available NPSH equals required NPSH. For this case the initial pressure is 12.8 psia which is the minimum Technical Specification value following implementation of containment conversion.



No plot is included for the Low Head Safety Injection (LHSI) pump since no COP is required in order to meet the NPSH requirements in the new analysis. The COP is required in the current analysis. The change in COP requirements for this pump is due to a change in the setpoint which signals the pump to switch to recirculation mode. This setpoint is being changed in the new analysis such that the LHSI pump will start at a later time when the sump level is higher and the sump temperature is lower. Both of these effects improve the NPSH available.

The following plots show the COP required for the Recirculation Spray pumps at BVPS-1 for both the current licensing basis analysis and the new analysis performed to support Containment Conversion and EPU. For the cases shown, the minimum pressure for the current analysis is 9.1 psia and for the new analysis the minimum pressure is 12.8 psia. These values correspond to the minimum allowable Technical Specification value for both current and post Containment Conversion conditions, respectively.





The maximum amount of time COP is required is indicated on these plots when the required COP reaches 0 psi. This value is approximately 400 seconds after the pumps start. This case represents the limiting case in the current analysis and a case in the new analysis which is appropriate for comparison. Other cases performed in the new analysis, e.g. minimum safeguards cases, show a COP required time which is longer due to a slower cooling of the sump with minimum heat removal systems operating. The maximum time COP is required is approximately 20 minutes for all cases. These cases were not analyzed in the current analysis since they are non-limiting with respect to NPSH margin; however, similar trends would be expected.

It can also be seen that the new analysis increases this time requirement by about 40 seconds relative to the current analysis. The plots also show that the COP required in the new analysis increases by approximately 2 psi over the current analysis. Considering the numerous input changes and a change in the computer program used to perform these analyses (i.e., MAAP vs. LOCTIC), the two curves demonstrate that the changes in COP required are relatively insignificant.

It should be noted that BVPS-1 is licensed to a methodology for calculating NPSH as opposed to a specific value of COP. The NRC acceptance to this approach is documented in a letter dated March 26, 1999, which discusses completion of licensing actions for Generic Letter 97-04. The methodology used for calculating COP and NPSH for the analysis supporting containment conversion and EPU is identical to that approved in the referenced letter. This approach is consistent with the methodology discussed in paragraph 4 of Section 1.3.1.1 in the draft Revision 4 to Regulatory Guide 1.82.

Question:

3.a. Do the values of required NPSH remain constant for the duration of the postulated LOCAs?

Response:

The values of required NPSH remain constant for the duration of the postulated LOCA. The required NPSH only changes as a function of the pump flow. Both RS and LHSI flows are essentially constant in all modes of operation; i.e., injection mode, cold leg recirculation and hot leg recirculation.

Question:

3.b. Are the EPU and pre-EPU required NPSH values the same for the ECCS and containment heat removal pumps?

Response:

The EPU will not result in any changes to the ECCS (LHSI) or containment heat removal (RS) pumps. The required NPSH values are based on pump testing and are a function of pump flow. The required NPSH will not change since no changes to the pumps are taking place and the flow rates for EPU are essentially the same as pre-EPU conditions.

Question:

3.c. What conservatism is in the required NPSH values?

Response:

No credit is taken for a reduction in required NPSH at temperatures higher than those used to establish the required values. The required NPSH values are based on specific pump tests by the manufacturer and correspond to the standard definition (3% reduction in total developed head). Testing of the BVPS-1 RS pumps has shown that the pumps can operate in a stable condition at NPSH values below the standard level. These tests are discussed in the response to Question 7 of this enclosure.

Question:

4. If containment accident pressure is needed in order to ensure that ECCS pumps have adequate NPSH under "best-estimate" conditions, then a transient or other initiating event and a failure of containment could lead to cavitation and failure of the ECCS pumps. The failure of containment could be from a pre-existing hole or from a failure of containment isolation, for example. This failure mode of the ECCS pumps is not typically modeled in PRAs.

Increasing power to EPU conditions may result in a longer duration of the need for crediting containment accident pressure, which in turn could increase the risk of core damage scenarios where ECCS pumps fail because of loss of NPSH. If no credit for containment accident pressure is required, post-EPU, under "best-estimate" assumptions, then it could be argued that there would be no risk from this scenario. Similarly, if the ECCS pumps could be shown to not fail because of cavitation for the period of time that containment pressure would be needed to provide adequate NPSH, again it could be argued that this scenario posed no risk.

Does the EPU result in any change in risk caused by a change in the amount or duration of containment accident pressure credit necessary to demonstrate adequate NPSH for any ECCS pump? If "no," please provide the basis for this conclusion. If "yes:"

a. Describe how containment accident pressure credit is modeled in the probabilistic risk assessment model, or provide the basis for not including it.

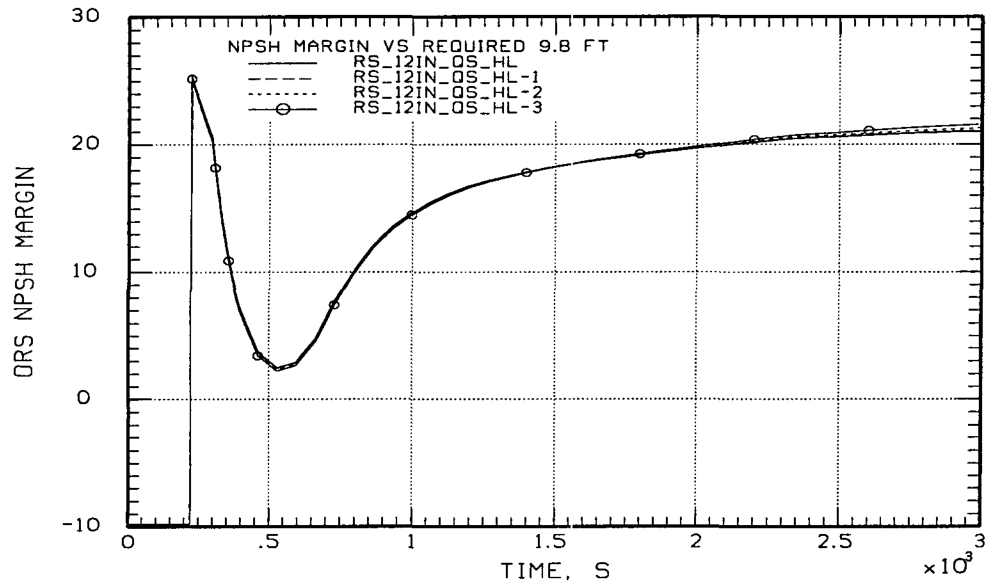
b. What is the change in risk impact of crediting containment accident pressure for ensuring adequate NPSH for the ECCS pumps that is expected from the EPU? This should be expressed in terms of the change in core damage frequency and large early release frequency. The risk evaluation should include consideration of uncertainty.

c. Describe how crediting containment accident pressure in this manner impacts defense-in-depth and safety margins. Describe the performance measurement strategies that will ensure the containment and affected ECCS pumps will be monitored commensurate with their safety importance, that feedback of information and corrective actions will accomplished in a timely manner, and that degradation in performance will be detected and corrected before plant safety can be compromised.

Response:

Based on the assessment provided in the response to Question 3 regarding the amount and duration of required Containment Overpressure (COP), there is an insignificant change in risk resulting from EPU with respect to crediting COP. The required COP is of relatively short duration (10-20 minutes) and margin exists to the available COP. The available COP (and NPSH) is conservatively calculated assuming all parameters are biased simultaneously in the conservative direction. Testing of the RS pumps as described in the response to Question 7 has demonstrated that the pumps are capable of stable operation at conditions where NPSH is reduced below the standard requirement.

To further demonstrate that there is an insignificant risk impact and that the current PRA model adequately accounts for COP requirements, a study was done to determine the impact of operation of the RS pumps under accident conditions with failures of containment isolation for systems which communicate directly with the containment atmosphere. The largest such penetration at BVPS-1 is two (2) inches in diameter. The estimated frequency of a containment isolation failure coincident with a large LOCA is approximately 1.0 E-8. The following plot shows the results on NPSH margin for the limiting ORS pump case considering containment isolation failures for a range of sizes of one (1) inch through three (3) inches in diameter. The IRS pump results are bounded by the ORS pump results. The results show that these failures have an insignificant impact on the calculated NPSH margin.



On the basis of the preceding discussions, it is concluded that the change in risk associated with EPU with respect to COP is insignificant and the risk associated with crediting COP at BVPS-1 is low enough not to be considered in the PRA model.

Question:

5. Describe indications available to the operator of a cavitating ECCS or containment heat removal pump.

Response:

Indications of pump motor amps and discharge pressure or flow are available to the operator to directly monitor for cavitation. There are steps in the Emergency Operating Procedures which direct the operator to monitor for pump cavitation and the operators are trained to observe these indicators for signs of cavitation.

Question:

5.a. What actions are available to the operator to restore adequate available NPSH if cavitation is detected?

Response:

In the event cavitation is indicated, the operator is instructed to shut down the running pump and attempt to restart the pump when there is a higher water level in the containment sump. Additionally, contingency procedures have been put in place to conserve water available for injection from the RWST in the event that a failure of recirculation occurs due to debris clogging or some other event which prevents flow from the containment sump.

Question:

6. With realistic (nominal) assumptions (nominal power, decay heat, pump flows, no single failure, nominal service water temperature, etc.), is credit for containment accident pressure necessary for adequate available NPSH for the ECCS and containment heat removal pumps?

Response:

Nominal or realistic assumptions do not alleviate the need for crediting COP. This is principally due to the fact that none of the parameters which can be changed to more realistic values directly or significantly affect the sump temperature when the RS pumps start. At BVPS-1, the RS pumps start approximately 4 minutes following a high-high containment pressure signal. These pumps draw water directly from the sump at this point. At this point in the transient, the sump level is relatively low and the temperature is high. The inventory in the sump is made up primarily of spilled RCS and, at this point, several minutes of containment spray. Changes in inputs parameters which can be adjusted to nominal conditions do not provide enough reduction in sump temperature to provide acceptable results without COP. In order to eliminate the need for COP, a change in the elevation of the pump would be required, which is impractical.

Question:

7. Describe any experience of Beaver Valley ECCS or containment heat removal pumps operating under cavitating conditions. Describe the duration of any tests and the results.

Response:

Experience with BVPS ECCS and containment heat removal pumps is based on specific pump tests which were conducted in 1977. The purpose of these tests was to determine the possible effect on pump head and flow at reduced NPSH, determine the performance under cavitation, and demonstrate that after operation at reduced NPSH, the pumps would perform in a normal manner after full restoration of NPSH. These tests were performed for both the RS and LHSI pumps. The tests demonstrated that for short term operation (less than 1 hour), stable operation at reduced performance could be expected with no damage to the pumps.

This testing was discussed in a report submitted to the NRC in September 1977. The information was captured in an SER dated September 30, 1977. The SER provided the following:

"Copies of the following documents are available for public inspection in the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. 20555 and at the Beaver Area Memorial Library, 100 College Avenue, Beaver, Pennsylvania, (1) Letters from DLC dated August 20, 1977, August 25, 1977 September 8, 1977 and September 19, 1977, (2) Stone and Webster Report entitled "Analysis and System Modification for Recirculation Spray and Low Head Safety Injection Pumps Net Positive Suction Head", dated September 9, 1977, and (3) this Order for Modification of License, In the Matter of Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, Beaver Valley Power Station Unit No. 1, Docket No. 50-334".

Question:

8. What single failure is assumed for the available NPSH calculations?

Response:

Postulated single failures are considered in establishing the limiting available NPSH cases. A sensitivity study was performed to determine the limiting single failure as well as the direction of bias for other input parameters. The limiting single failure for the Recirculation Spray (RS) pumps is a failure of one Quench Spray pump. The limiting single failure for the LHSI pump is a failure of one emergency diesel generator.

Question:

8.a. If no single failure is assumed, is credit for containment accident pressure necessary?

Response:

For the RS pumps, credit for containment accident pressure would still be required without an assumed single failure. The RS pumps start relatively early in a LOCA event while the sump water level is relatively low and the sump temperature is high. Generally, single failure assumptions do not have a significant impact on these parameters early in the accident.

Question:

9. What pump flow rates are assumed for the NPSH calculations?

Response:

The RS and LHSI pumps are modeled using pump and system curves. The actual calculated flow will depend on system alignment assumptions, number of pumps operating, and assumptions on pump performance (nominal or degraded). A typical range of values for the RS pumps is 3000-3500 gpm per pump. A typical range of values for the LHSI pumps is 2000-3400 gpm.

Question:

9.a. How do these assumed flow rates compare with instructions to the operator concerning pump flow in Emergency Operating Procedures for LOCAs?

There are no specific instructions in the Emergency Operating Procedures concerning pump flow in the RS or LHSI system. The systems are designed to operate without the need for flow control by the operators.

Question:

9.b. Are the pump flow rates assumed in the calculations maintained constant for the duration of the calculation?

The RS and LHSI pump flows are calculated based on changing system conditions. As stated previously, pump flows do not change substantially in the different modes. In each mode the pump flows will be relatively constant.

Question:

9.c. Can ECCS or containment heat removal pump flow be throttled?

The RS system is not designed to be throttled. The LHSI system contains cavitating venturis which limit flow to the maximum assumed in the NPSH analysis and no throttling is required.

Additional Questions regarding MAAP analysis for SBO:

On December 9, 2005, the licensee replaced its previous answer to RAI question 2.c. That question was:

Table 10.16-1 gives pre- and post-EPU times to core damage for station blackout scenarios. Why does this time increase on BVPS-1 and decrease on BVPS-2 for the "182 gpm, successful cooldown/depressurization, primary plant demineralized water storage tank make-up available" case?

Further clarification is needed to understand the results presented. For example, the Unit 1 pressurizer level increases from 9 hours to 17 hours post-EPU, but the accumulators appear to have finished injecting before 3 hours. As another example, the Unit 2 analysis shows a slightly shorter time to core damage post-EPU, whereas Unit 1 shows an increase. In order for the staff to determine whether this behavior is real or an attribute of the modeling, additional information is needed.

a. Please provide graphs of the following parameters as a function of time:

- **RCS pressure (cold leg)**
- **break mass flow rate**
- **two-phase level in the inner vessel (show the top and bottom elevation of the core)**
- **liquid level in the inner vessel**
- **secondary pressure vs time**
- **secondary ADV/PORV mass flow rate**
- **core exit steam temperature**
- **core normalized power**

b. Please provide results of the MAAP benchmarking efforts to demonstration that MAAP can compute time to uncovery for SBLOCAs. As such please provide the following benchmarks:

(1) Separate effects test comparisons to demonstrate MAAP can accurately predict two-phase level swell. Please show comparisons of MAAP to level swell tests with heat addition as well as depressurization experiments (see, for example, the THTF steady state uncovery tests as well as the 336 rod bundle uncovery tests in the G-2 facility, etc.).

(2) Also, please show comparisons of MAAP to SBLOCA integral system experiments that display long term core uncovery and heat-up of the fuel cladding to show that MAAP can predict the timing for uncovery of the core in scaled systems. See, for example, SEMISCALE Tests S-07-10D, S-U-T8, S-LH-1, S-LH-2, and ROSA-IV test SB-CL-01, etc.). Note: comparisons of MAAP to SBLOCAS that display no core uncovery are considered inadequate benchmarks.

Response:

Further clarification is being provided for the results presented in the FENOC Letter L-05-192 (Reference 1) for the response to RAI Question 2.c for the BVPS-1 pressurizer level increase from 9 to 17 hours for sequence SBO11 and the increase in the post-EPU core damage time.

In the BVPS PRA analyses, a number of station blackout (SBO) accident sequences are investigated. One of these (SBO11) considers a station blackout with auxiliary feedwater being available for an extended interval (approximately 24 hours) with the secondary side depressurized. After approximately 4.5 hours, the RCS is depressurized to approximately 300 psia and is held at this pressure for the next 25 hours. During the first 4.5 hours, accumulator injection refills the pressurizer to a level of approximately 4.5 feet. There is an injection from the accumulators just after 5 hours and this cold water initially reduces the boiled-up level in the RCS causing the pressurizer to almost empty. However, as this water temperature increases, the boiled-up level in the RCS is restored and causes the pressurizer level to increase approximately 7 feet. At approximately 6 hours, the pressurizer level increases by an additional 17 feet. The reason for this level increase has been investigated, and as discussed below, is due to the heat loss from the pressurizer to the containment when the surge line entrance is covered.

During a station blackout sequence, all of the pressurizer heaters are lost and therefore, early-on there is a net heat loss from the pressurizer to the containment. The heat loss to the containment for PWRs has been investigated through heat balances on plants, both in terms of a primary system heat balance as well as a containment heat balance. Table 1 "Data Sources for PWR Primary System Heat Losses" provides a summary of the results of the 1982 EPRI survey (Reference 4). These PWR systems tend to have heat losses from the RCS to the containment of approximately 2 MW or more. A value of 2 MW is used in the BVPS MAAP parameter file. It is noted that this heat loss is considerably greater than the through insulation heat loss and is attributed to the not-through insulation heat loss which is principally due to leakage through the insulation panels and penetrations. The MAAP model distributes this heat loss over the RCS surfaces and the resulting heat losses from the water and steam volumes in the pressurizer, under steady state full power conditions, are approximately 100 kW and 65 kW respectively. As the accident progresses, the heat loss to the containment changes as the temperature of the pressurizer wall changes; i.e. the heat loss decreases as the pressurizer wall temperature decreases. The energy transfer from the pressurizer water and steam inventories to the pressurizer wall is also determined by the wall temperature as well as the volumes of these media, which is determined by the sequence. Consequently, the energy transfer between the pressurizer wall and the water and steam may be from the steam and water to the wall, or the reverse, and can change during the transient. Nonetheless, the energy transfer between the pressurizer water and steam inventories is significant and has a nontrivial influence on the pressurizer level behavior.

Table 1
Data Sources for PWR Primary System Heat Losses
(Excluding Reactor Coolant Pump Motor Heat Load)

Plant	Design Power (MW_t - approx.)	Total Heat Loss (MW)	Data Source
Palisades	2300	1.1 - 1.4	Primary system heat balance*
Millstone-2	2600	2.5	Primary system heat balance*
CE-System 80	3800	4	Primary system heat balance*
Byron	3300	~ 2	Fan cooler heat balance less estimated Reactor Coolant Pump (RCP) motor heat load
McGuire	3500	2.2	Containment air flow and temperature-rise measurements

*Note: Generally, these data represent the results of tests where the RCP's were used to supply a constant heat input to the primary system. The heat rejection rate to the steam generators was measured by calculating how fast the steam generator levels changed with no feedwater addition while steaming at a rate that gave a constant primary system temperature. Alternatively, the feedwater flow rate required to keep the steam generator levels constant was measured. The difference between the pump heat and the heat rejected to the steam generators is the loss to the containment.

For SBO11, the Reactor Coolant System (RCS) is initially depressurized (to approximately 400 psi after 2 hours) such that there is reverse energy transfer from the pressurizer vessel wall to the lower temperature steam and water inside the pressurizer. During this interval, the pressurizer wall continues to lose energy to the containment. After approximately 5 hours, the heat loss from the pressurizer wall to the containment causes the wall temperature to decrease and eventually there is essentially no energy transfer between the pressurizer steam and water inventories and the pressurizer wall. Due to the continued heat loss to the containment, the wall temperature eventually decreases sufficiently that the energy transfer is from the steam and water to the pressurizer wall. This begins at approximately 6 hours and the average energy transfer rate from the steam region calculated by the code over the next three hours is approximately 32 kW. Meanwhile, the heat loss from the water region causes the temperature to decrease from the saturation value of approximately 432°F to approximately 408°F; i.e., the pressurizer water is subcooled. During this time, the RCS is held at essentially a constant pressure of 300 psi. A heat loss of 32 kW causes a condensation rate of 0.037 lbs/sec or a volumetric reduction rate of the steam region of 0.058 ft^3/sec. This heat removal rate and condensation continues for approximately 3 hours and corresponds to a steam volume decrease and a water volume increase of 628.5 ft^3, or a 14 foot increase in the pressurizer level. These calculations are close to the MAAP calculated water level increase and demonstrate that the reason for the pressurizer level to increase under these conditions is the combination of (a) the pressurizer inlet being submerged, and (b) the pressurizer volume experiencing a heat loss from the steam region to the pressurizer vessel wall and thus to the containment. Condensation of the steam causes water to be pulled from the RCS into the pressurizer. Also note that any steam pulled into the surge line is condensed by the subcooled water mass. This is a small addition to the water mass but more importantly, the incoming steam mass does not reach the steam region.

Eventually, the combined leakage rate from the three RCP seal LOCAs considered in this sequence results in sufficient voiding of the RCS that the surge line entrance is uncovered. When this occurs (approximately 10 hours into the sequence) the pressurizer starts to drain its inventory back into the RCS. Note that drainage does not occur at 17 hours as shown in Reference 1 but occurs at approximately 10 hours. Shortly after 10 hours, another accumulator injection occurs. This change in the on-set of pressurizer draining is due to an error in the BVPS MAAP parameter file associated with the pressurizer surge line configuration; i.e., the inclusion of a loop seal in the surge line and the assumption that the surge line enters from the top of the hot leg. The BVPS units do not have a loop seal and the surge line enters at the center line elevation of the hot leg. However, the error associated with the surge line configuration does not impact MAAP results that occur prior to the hot leg and attached surge line being uncovered, as inventory is lost from the RCS. The post-EPU pressurizer level provided for the SBO11 sequence in Reference 1 included the effect of a loop seal such that the elevated level (approximately 20 ft) was sustained until approximately 17 hours when the surge line entrance was uncovered. The revised graphical information discussed below uses the proper surge line configuration; i.e., no loop seal and the surge line enters at the centerline of the hot leg.

The remaining MAAP SBO sequences were also re-evaluated as a result of the change in the BVPS MAAP parameter file associated with the pressurizer surge line configuration. Based on the results of this re-evaluation, it was determined that core damage times for some of the SBO sequences were slightly impacted. To address these changes in core damage times, the SBO electric power recovery models were re-evaluated using the revised MAAP output. Preliminary results show that there is no significant impact on CDF at either unit (i.e., less than 2.1E-08 increase) due to the changes in the electric power recovery model.

To determine the impact of the surge line configuration error on the HRA and success criteria, all other remaining MAAP cases are being re-analyzed. Preliminary results show that there are slight changes to some of the operator action timings, but not so significant as to impact the performance shaping factors. Therefore, the HEPs are not expected to change at either unit as a result of the error associated with the surge line configuration. In addition, no changes to the success criteria are expected.

The effects of the changes to the BVPS MAAP parameter file associated with the pressurizer surge line configuration are not expected to significantly impact the results reported in the EPU LAR and associated RAI responses. Confirmation of these re-analyses results will be included in the supplemental response that will address Questions 1 and 2 of this enclosure.

In Table 2-7 of Reference 1, the post-EPU time to core damage for BVPS-1 increased (30.6 hours and 30.3 hours, with seal binding failure at 30 minutes and 13 minutes, respectively) compared to the pre-EPU core damage time of 27.0 hours; unlike BVPS-2 that showed a decrease in core damage time for the uprated power condition. This increase in the BVPS-1 post-EPU core damage time was mostly attributed to changes in the Technical Specification accumulator water mass and minimum pressure between the pre-EPU and post-EPU conditions, as discussed in Reference 1.

To further investigate this phenomena, an additional sensitivity case was run by modifying the pre-EPU MAAP model with the post-EPU accumulator parameters (mass, pressure, and volume) listed in Table 2.8 of Reference 1. The results of this analysis showed the time to core damage increased from the 27.0 hours reported for the pre-EPU case in Table 2.7, to about 32.1 hours. Since this sensitivity time is about 1.5 hours longer than the 30.6 hours reported in Table 2.7 of Reference 1 for the post-EPU model with seal binding failure at 30 minutes, it reconfirms that there is a strong correlation between the accumulator parameters and the time to core damage. It also indicates that when using the same initial post-EPU accumulator parameters, the post-EPU time to core damage occurs earlier than with pre-EPU conditions, which is what was expected (and similar to BVPS-2) due to the increased core decay heat levels. The lesser impact of other changes including a slightly larger primary system mass associated with the replacement steam generators were also discussed in Reference 1.

a. Additional Graphic Information

Additional graphical information regarding the SBO11 sequence and a similar sequence without depressurization (SBO13) is provided in Attachment A "Graphical Information for BVPS-1 Sequences SBO11 and SBO13" of this enclosure. A brief discussion of this additional graphical information follows.

The definitions of the station blackout sequences are provided in Table 2 "Sequence Attributes for SBO Scenarios". The difference is that SBO11 includes steam generator cooldown and depressurization while SBO13 does not. However, SBO13, like SBO11, does include successful auxiliary feedwater (AFW) which prevents rapid steam generator dryout. Thus, the core damage time for SBO13 occurs at 8.1 hours, which is several hours later than a similar sequence (SBO15) that does not credit AFW. The SBO15 sequence does not credit AFW and it resulted in a core damage time of 2 hours.

Table 2
Sequence Attributes for SBO Scenarios

Sequence Attributes	SBO11	SBO13
BVPS-1 station blackout (SBO) with reactor coolant pump (RCP) seal Loss of Coolant Accidents (LOCAs) in all loops.	Yes	Yes
Initial seal leak rate is 21 gpm per loop.	Yes	Yes
Seal leak rate is increased to 182 gpm per loop at 13 minutes due to assumed seal binding failure.	Yes	Yes
Successful turbine driven AFW feeding a single (broken loop) steam generator with unlimited battery capacity.	Yes	Yes
Successful Primary Plant Demineralized Water Storage Tank (PPDWST) refill.	Yes	Yes
Successful steam generator cooldown and depressurization using a single (broken loop) steam generator atmospheric dump valve (ADV) to obtain cooldown at maximum rate. Cooldown may exceed 100°F/hr as ADV is full open for duration of depressurization. Cooldown and depressurization begins at 30 minutes.	Yes	No
2 of 3 cold leg accumulators are available.	Yes	Yes
Sequence end time is the time of reactor pressure vessel (RPV) failure.	Yes	Yes

In SBO11, the RCS pressure drops below the accumulator pressure in a few hours such that partial accumulator injection occurs. In SBO13, the RCS pressure remains above the accumulator pressure for approximately 8 hours so no accumulator injection occurs during the 8 hour interval. In both sequences the pressurizer initially drains within an hour or less of accident initiation but then partially refills as void forms in the RCS and the surge line is again covered. Subsequently, the surge line is again uncovered due to inventory loss through the seal LOCAs and the pressurizer drains again. For SBO11, the final pressurizer draining is delayed until approximately 12 hours since partial accumulator injection has occurred before this time. For SBO13, the final pressurizer draining occurs earlier (approximately 5 hours) as no accumulator injection has occurred by that time.

b. MAAP Benchmarking Information

The following MAAP benchmarking results are provided to demonstrate that MAAP can reliably compute the time to uncover the core for small LOCAs and other postulated transients.

Comparisons of the MAAP liquid swell model with separate effects tests demonstrate MAAP realistically represents the two-phase level swell. These benchmark results have been documented as a draft section for the separate effects benchmarks in Volume 3 of the MAAP4 Users Manual, "Benchmarking". The details of these separate effects benchmarks are provided in Attachment B "Benchmarking Results" from MAAP4 - Separate Effects VFVOL, a copy of which is attached to this enclosure. The pool average void fraction for a boiling water pool with volumetric heating is calculated by the function VFVOL in MAAP4. Attachment B discusses the separate effects function (VFVOL) and its comparison with (a) the THTF steady state uncovery tests, (b) the in-reactor Full Length High Temperature (FLHT) experiment 4 results and (c) the INER integral system tests (IIST) measurements of the water swell (boilup) for a small break LOCA test.

The MAAP4 benchmark comparisons with scaled thermal-hydraulic experiments, plant events and the TMI-2 accident that includes long term core uncovery and heat-up of fuel cladding are available in Volume 3 of the MAAP4 User's Manual (Reference 2). Benchmarks with the Semiscale Tests S-LH-1 and S-LH-2 have been performed with the MAAP3B computer code that uses a liquid swell model that is the same as that used in MAAP-DBA. These Semiscale Tests benchmarks have been documented in EPRI Report TR-100741 (Reference 3). An additional MAAP4 benchmark for an integral station blackout experiment performed in the IIST facility has also been performed and documented in a paper presented at NURETH-11 (Reference 5). More details of these comparisons are available in these references. A brief summary of predicted and experimental times that the core was uncovered is provided below and reasonable agreement is shown.

	Time Core Uncovered (seconds)	
Experiment	**MAAP**	**Experiment**
Semiscale S-LH-1	380	420
Semiscale S-LH-2	380	470
IIST	10,700	11,700

References

1. FENOC Letter L-05-192, Supplemental PRA Information in Support of License Amendment Request Nos. 302 and 173, Extended Power Uprate (EPU), dated December 9, 2005.

2. EPRI, 2004, MAAP4 Users Manual, Report prepared by Fauske & Associates for EPRI.

3. EPRI, 1992, EPRI TR-100741 (Project 3044-01), "MAAP Thermal Hydraulic Quantification Studies," June.

4. EPRI, 1982, "Control of Containment Air Temperature: An Industry Survey and Insulation Test," EPRI Report NP-2694.

5. Henry, R. E., et al., 2005, "Comparison of the MAAP4 Code with the Station Blackout Simulation in the IIST Facility," 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-11), Avignon, France, October 2-6.

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Graphical Information for BVPS-1 Sequences SBO11 and SBO13

Plots for the following variables are provided for BVPS-1 for the two station blackout sequences (MAAP Run U1_SBO11 and MAAP Run U1_SBO13):

- RCS and secondary (SG) pressures
- Break mass flow rate
- Two-phase level in RCS
- Liquid level in RCS
- Secondary ARV steam flow rate
- Core exit steam temperature
- Core normalized power
- Pressurizer level
- Accumulator water mass

(Note: The two-phase and liquid level plots do not start at time zero. They start once the phases have separated in the RCS.)

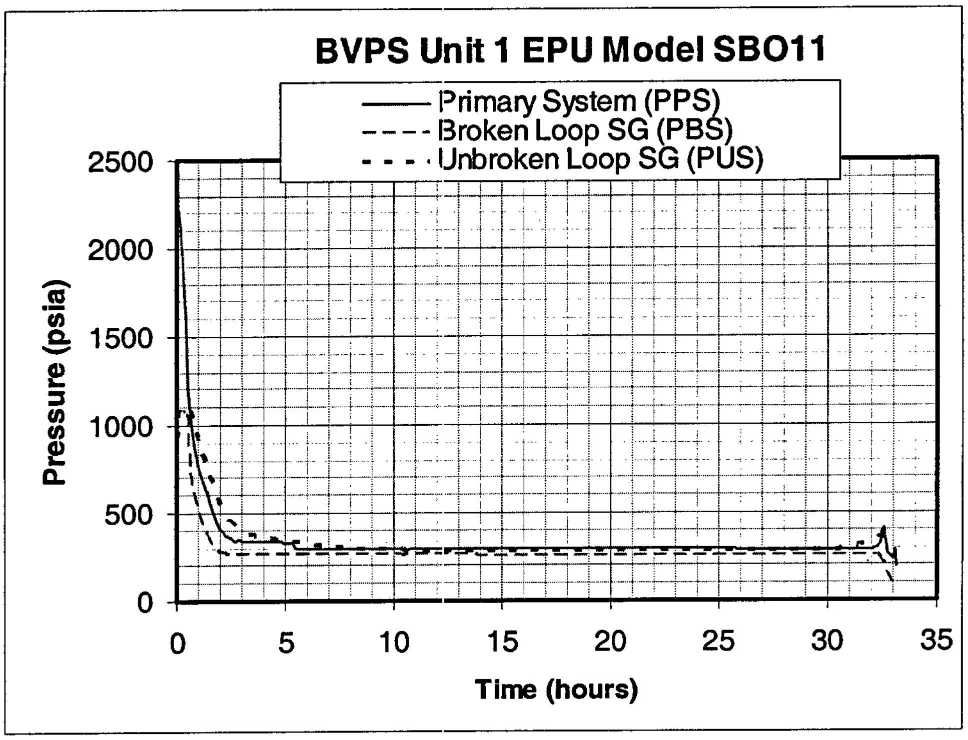


Figure 1 RCS and Steam Generator Pressures for BVPS-1 SBO11

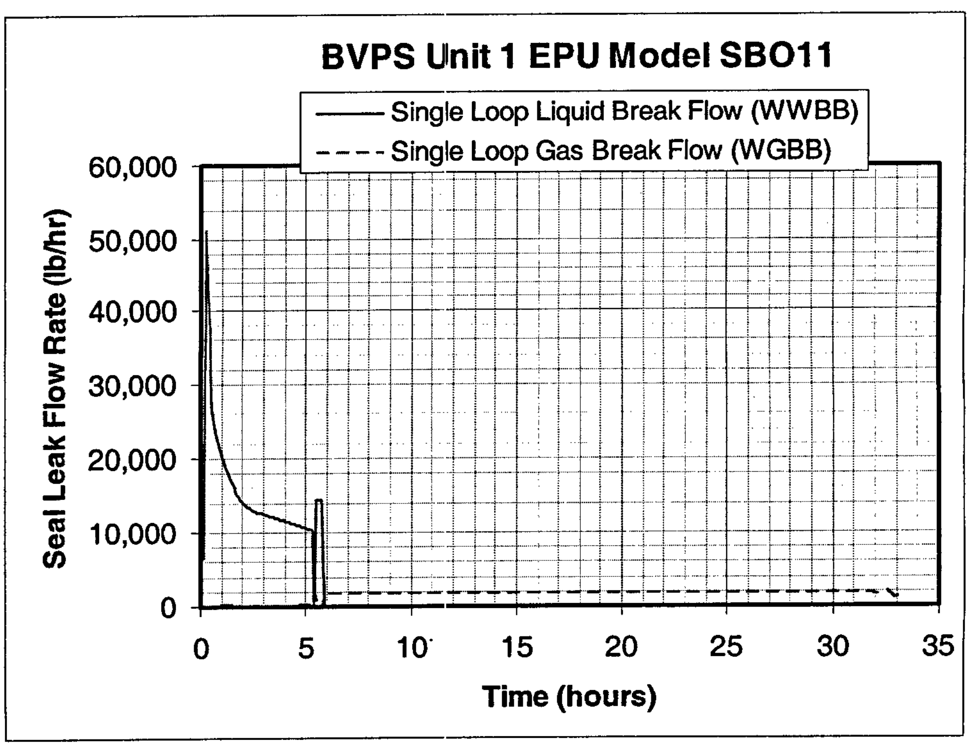


Figure 2 Break Mass Flow Rate Due to an RCP Seal Leak Flow Rate for a Single Loop for BVPS-1 SBO11 (Identical seal leak exists in all three loops)

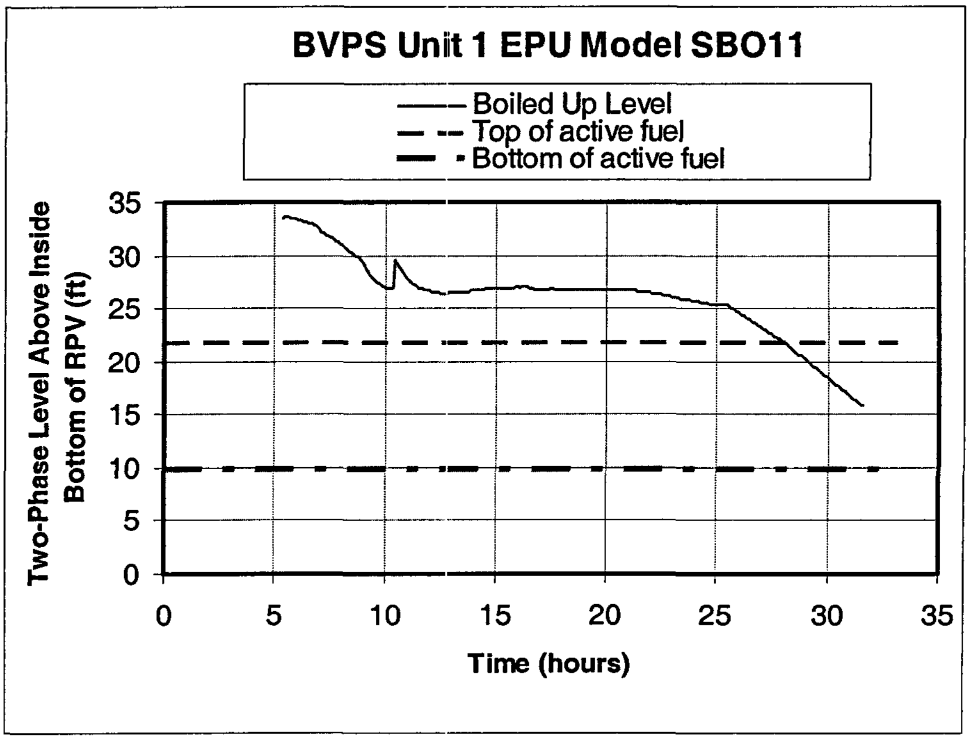


Figure 3 RPV Two-Phase Level for BVPS-1 SBO11

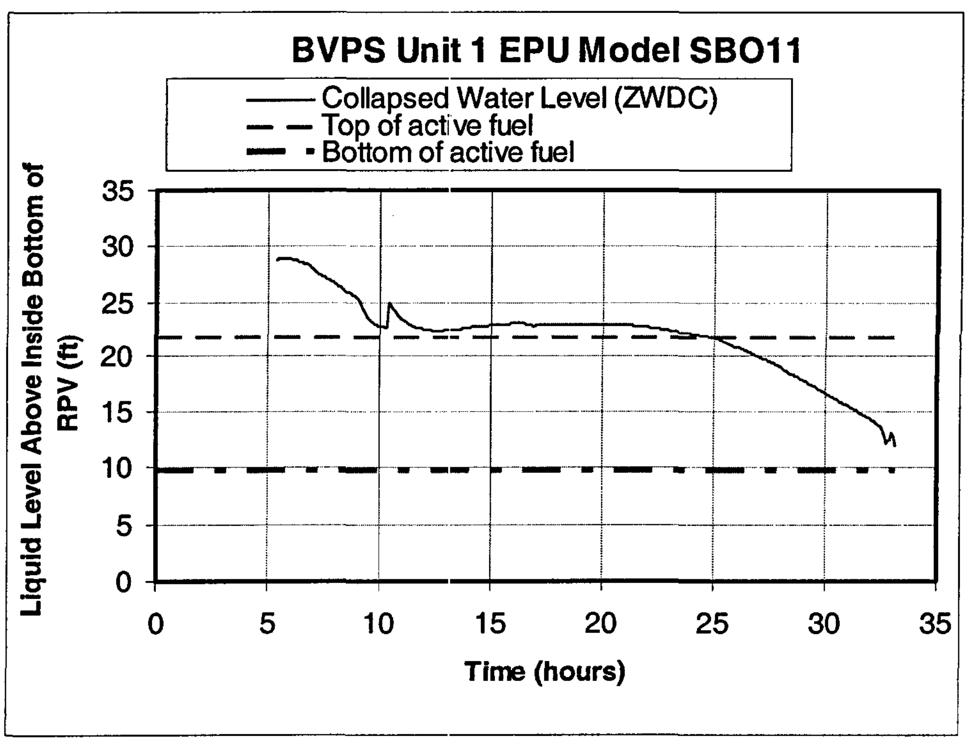


Figure 4 RCS Liquid Level for BVPS-1 SBO11

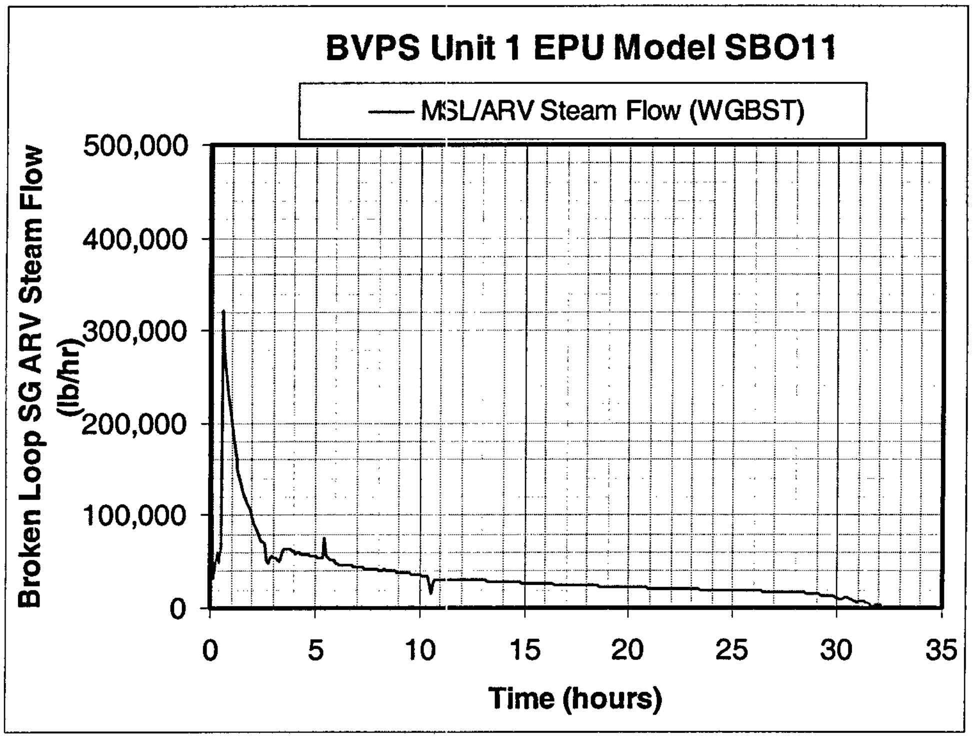


Figure 5 Secondary ARV Steam Flow for a Single Steam Generator for BVPS-1 SBO11 (Cooldown and depressurization performed with a single steam generator)

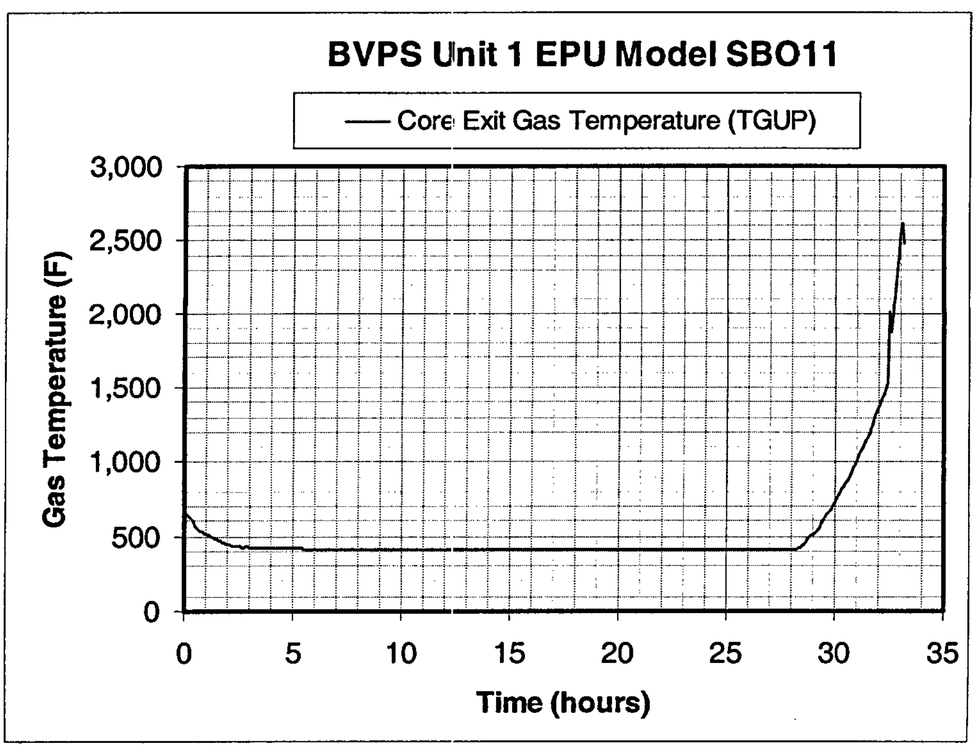


Figure 6 Core Exit Steam Temperature for BVPS-1 SBO11

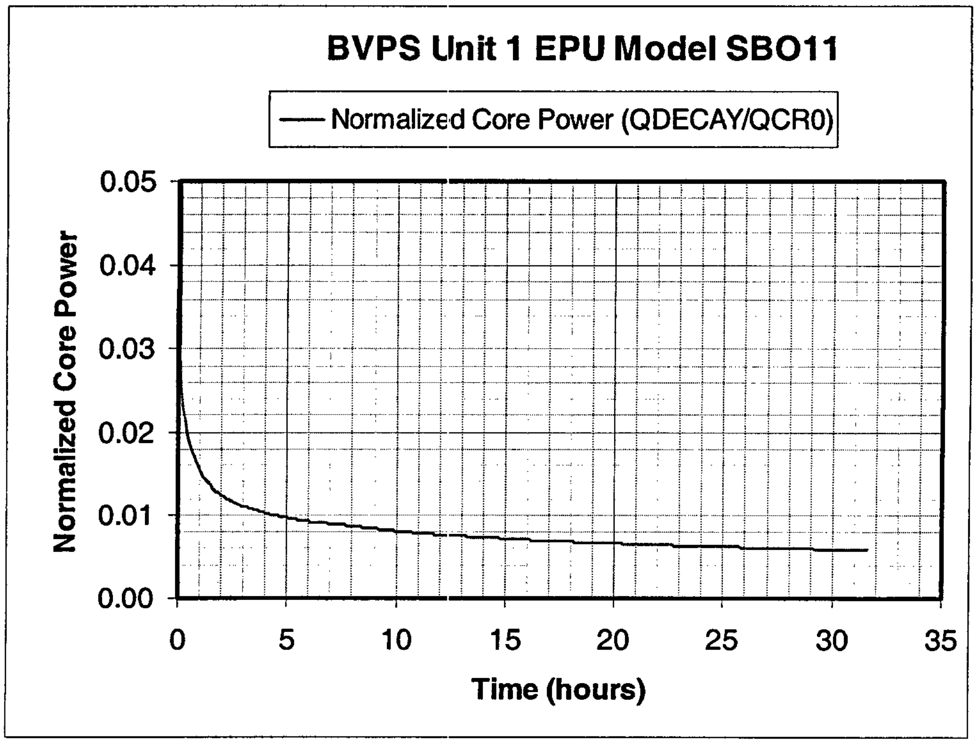


Figure 7 Core Normalized Power for BVPS-1 SBO11

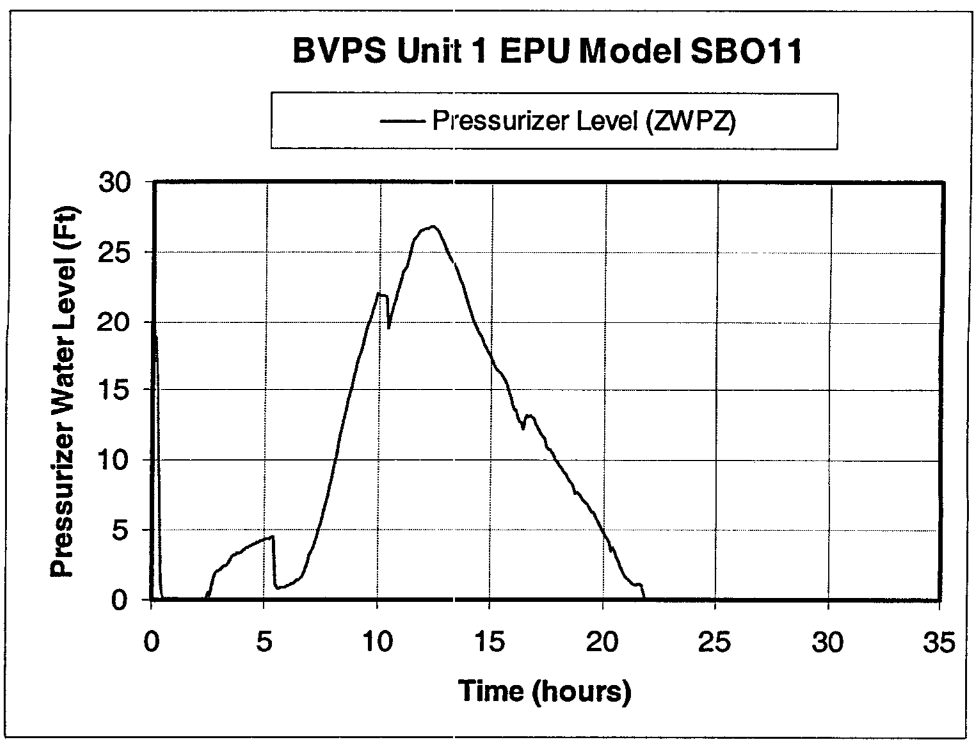


Figure 8 Pressurizer Level for BVPS-1 SBO11

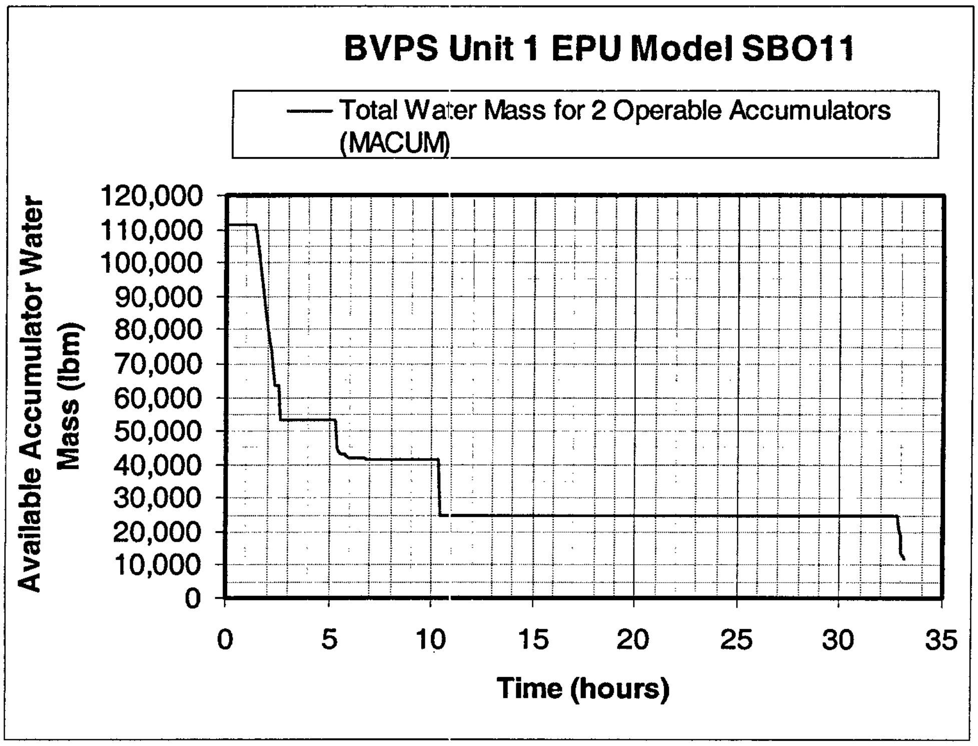


Figure 9 Total Accumulator Water Mass for 2 Operable Accumulators for BVPS-1 SBO11

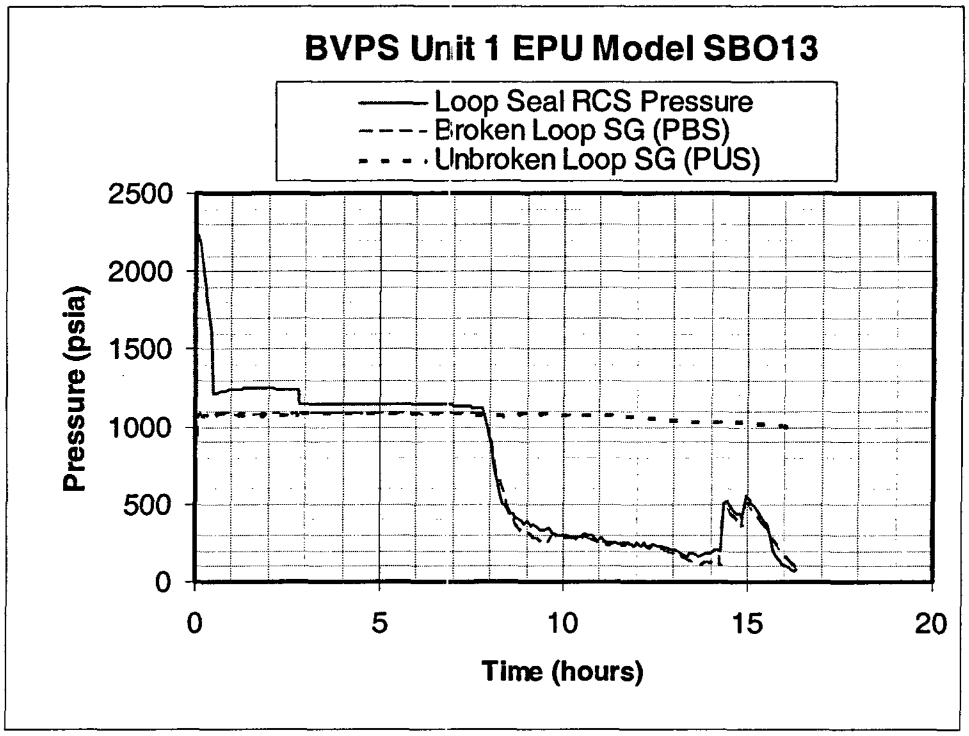


Figure 10 RCS and Steam Generator Pressures for BVPS-1 SBO13

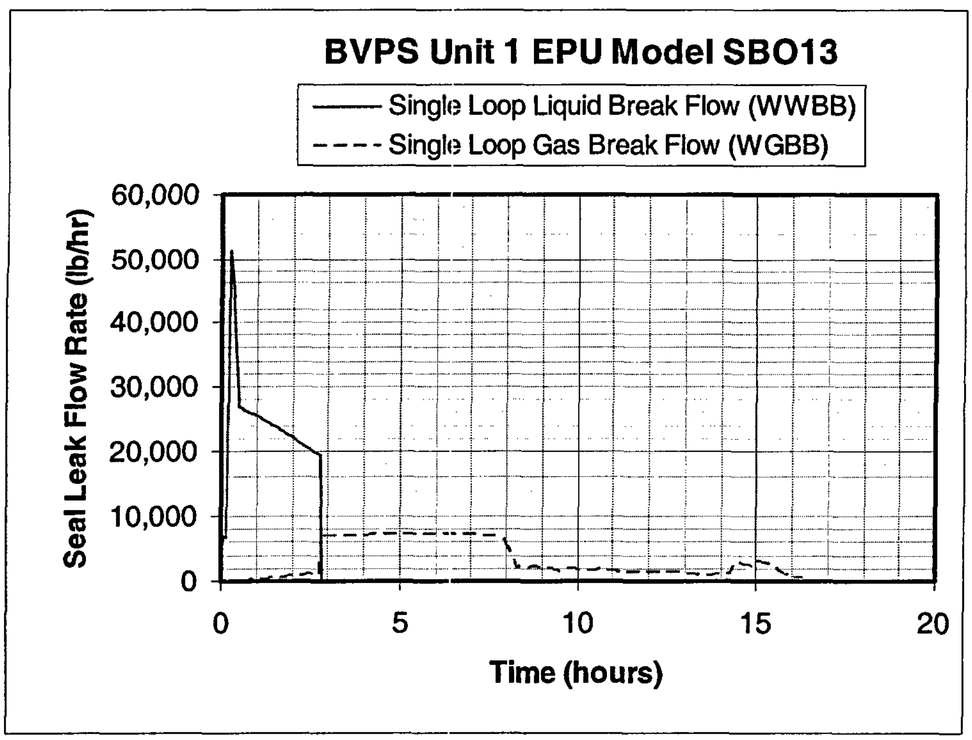


Figure 11 Break Mass Flow Rate Due to an RCP Seal Leak Flow Rate for a Single Loop for BVPS-1 SBO13 (Identical seal leak exists in all three loops)

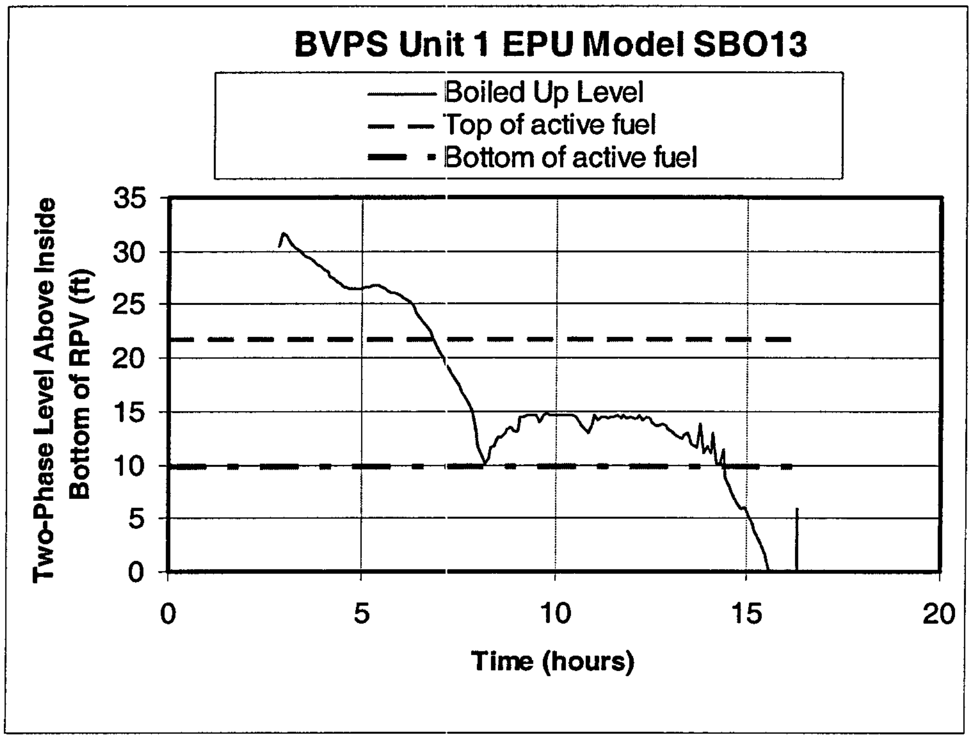


Figure 12 RPV Two-Phase Level for BVPS-1 SBO13

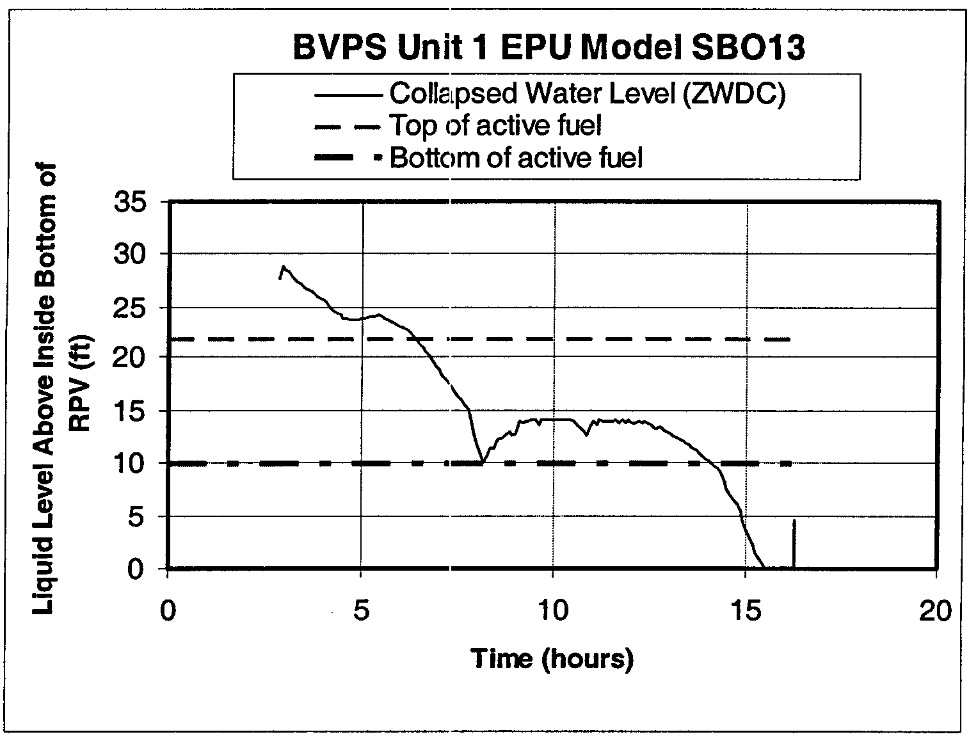


Figure 13 RCS Liquid Level for BVPS-1 SBO13

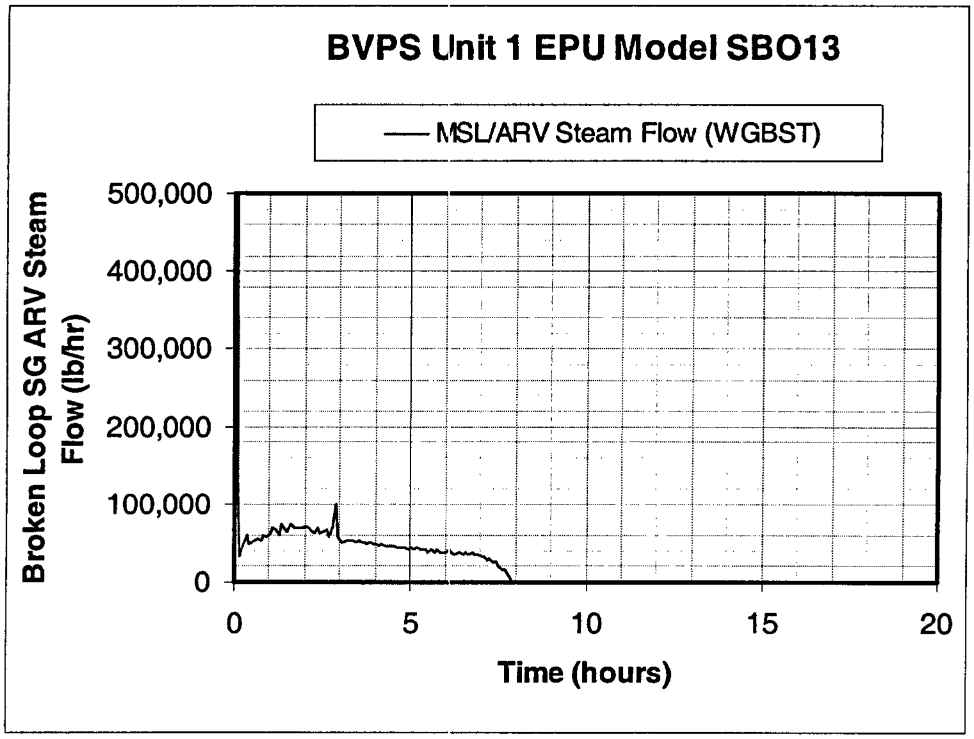


Figure 14 Secondary ARV Steam Flow for a Single Steam Generator for BVPS-1 SBO13

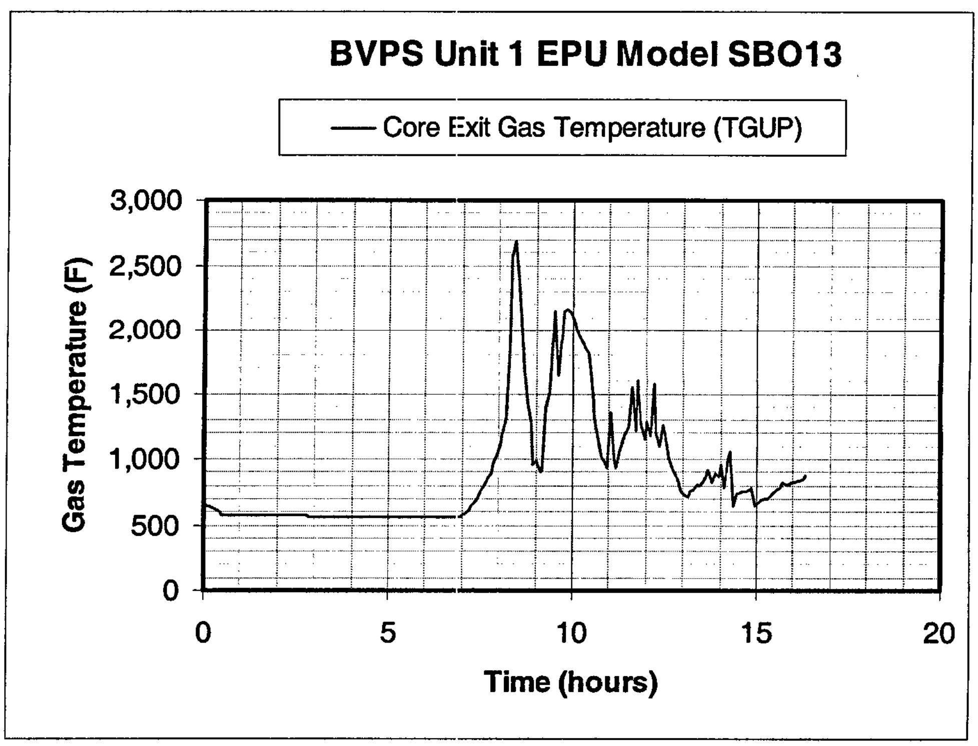


Figure 15 Core Exit Steam Temperature for BVPS-1 SBO13

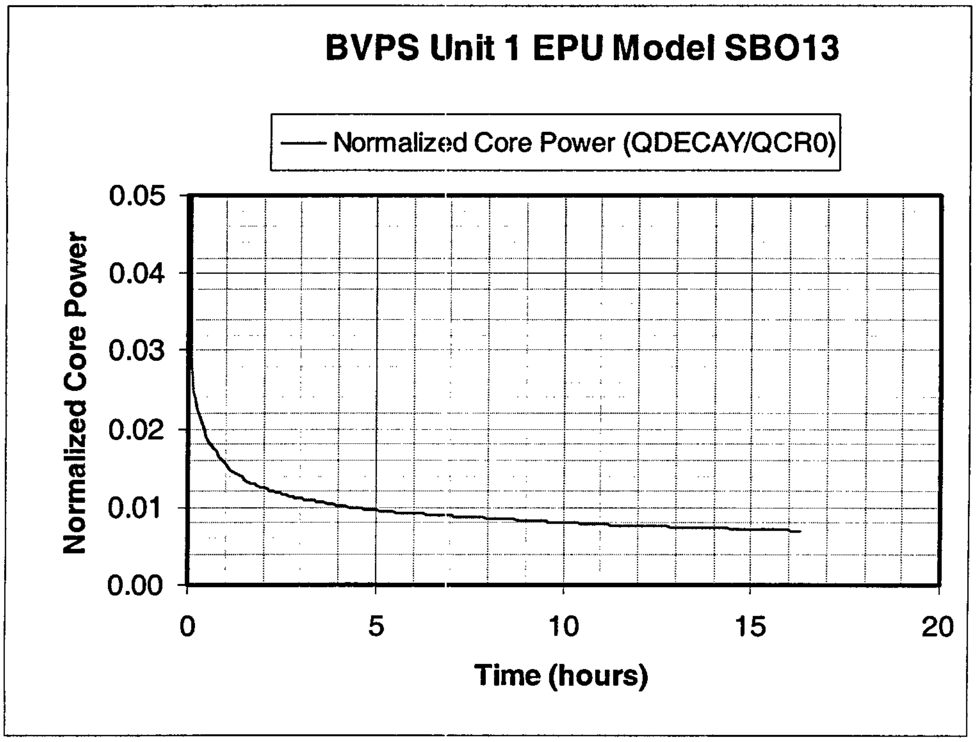


Figure 16 Core Normalized Power for BVPS-1 SBO13

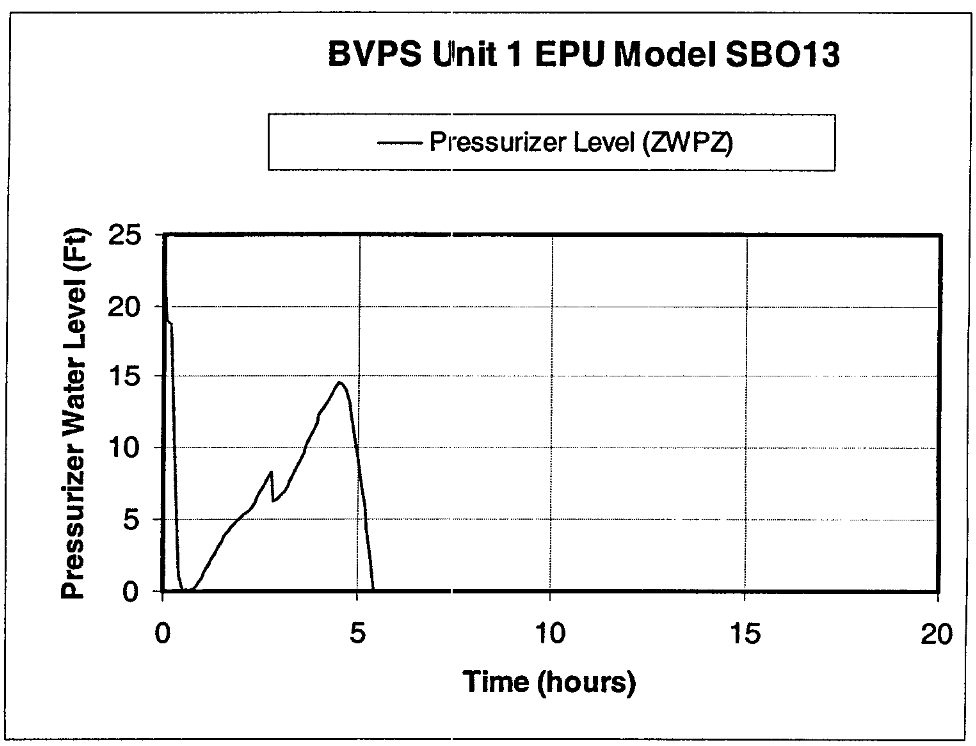


Figure 17 Pressurizer Level for BVPS-1 SBO13

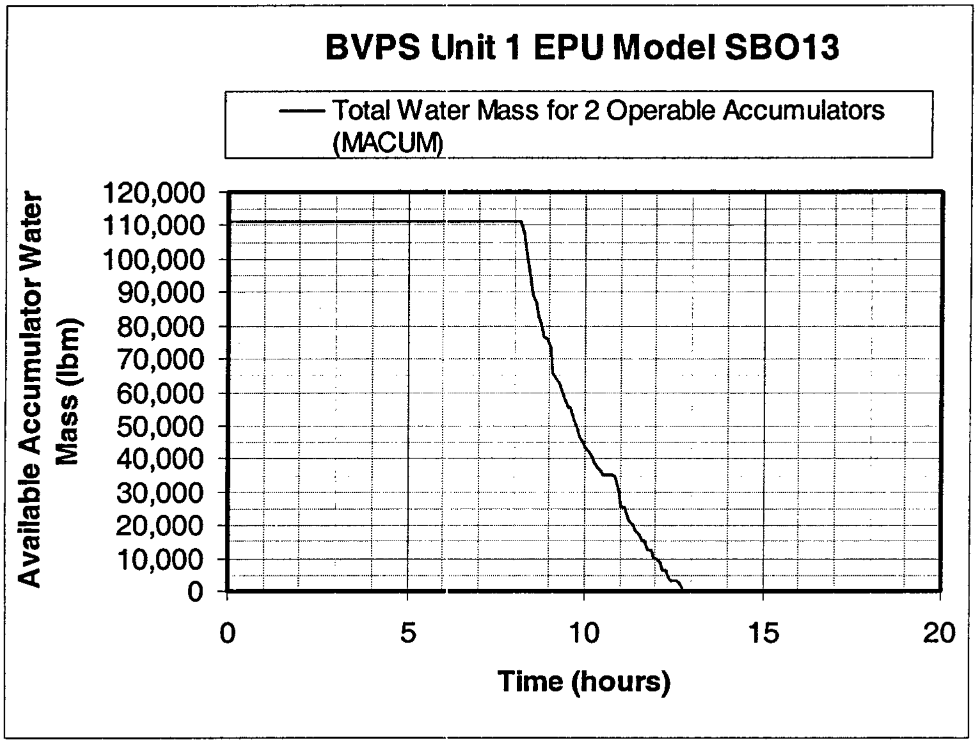


Figure 18 Total Accumulator Water Mass for 2 Operable Accumulators for BVPS-1 SBO13

Attachment B of Enclosure 1 of L-06-003

Benchmarking Results

BOILED-UP LIQUID LEVEL

1.0 INTRODUCTION

When analyzing the response of a reactor core to possible accident conditions, the boiled-up water level is a very important phenomenon. The boiled-up level maintains cooling of the nuclear fuel pins even when the collapsed water level is below the Top of Active Fuel (TAF). Specifically, as the water level decreases in the Reactor Pressure Vessel (RPV), boiling in the reactor core causes the water level to be higher than the collapsed water level in the downcomer region around the reactor core. This higher level has been called many names such as (a) the boiled-up level, (b) the level swell and (c) liquid froth height.

Because of its importance under accident conditions, several experiments have been performed to investigate the fundamental nature of the phenomenon. Some of these were conducted in separate effects tests in a fuel bundle configuration. Two of these experimental programs are the Oak Ridge National Laboratory test performed in the Thermal Hydraulic Test Facility (THTF) (Anklam et al., 1982) and the in-reactor Full-Length High Temperature (FLHT) experiments reported by the Pacific Northwest Laboratory (PNL) (Lanning et al., 1983). Other sources of information are the integral experiments such as the IIST scaled facility described by Lee et al. (1999).

2.0 BOUNDARY CONDITIONS

In the MAAP4 code, the function VFVOL calculates the average void fraction below the boiled-up level based on the superficial steam velocity produced by boiling in the core. Therefore, to perform the benchmark, one only needs to know the steam generation rate and to a limited extent the system pressure. References for these experiments either give these parameters explicitly or provide sufficient information for them to be calculated.

2.1 THTF Tests

Figure 1 depicts the THTF in the configuration used for the small break tests. Coolant flow is circulated through the vertical test section using the steady-state inlet flow line. An isolated depiction of the vertical test section is given in Figure 2, note that this is a full height facility. Figure 3 shows the cross-sectional view of the annular shroud, the bundle shroud box and the 64 positions in an 8 x 8 configuration. Four of these locations have unheated rods as illustrated in Figure 4. For the readers reference, Figure 5 shows the instrument positions over the full height (12 foot heated length) test facility.

A set of steady-state mixture level swell tests were performed in the THTF facility and the results are listed in Table 1. As listed in the table both the boiled-up and collapsed levels were

Figure 1: THTF in Small Break Test Configuration.
(Taken from Anklam et al., 1982.)

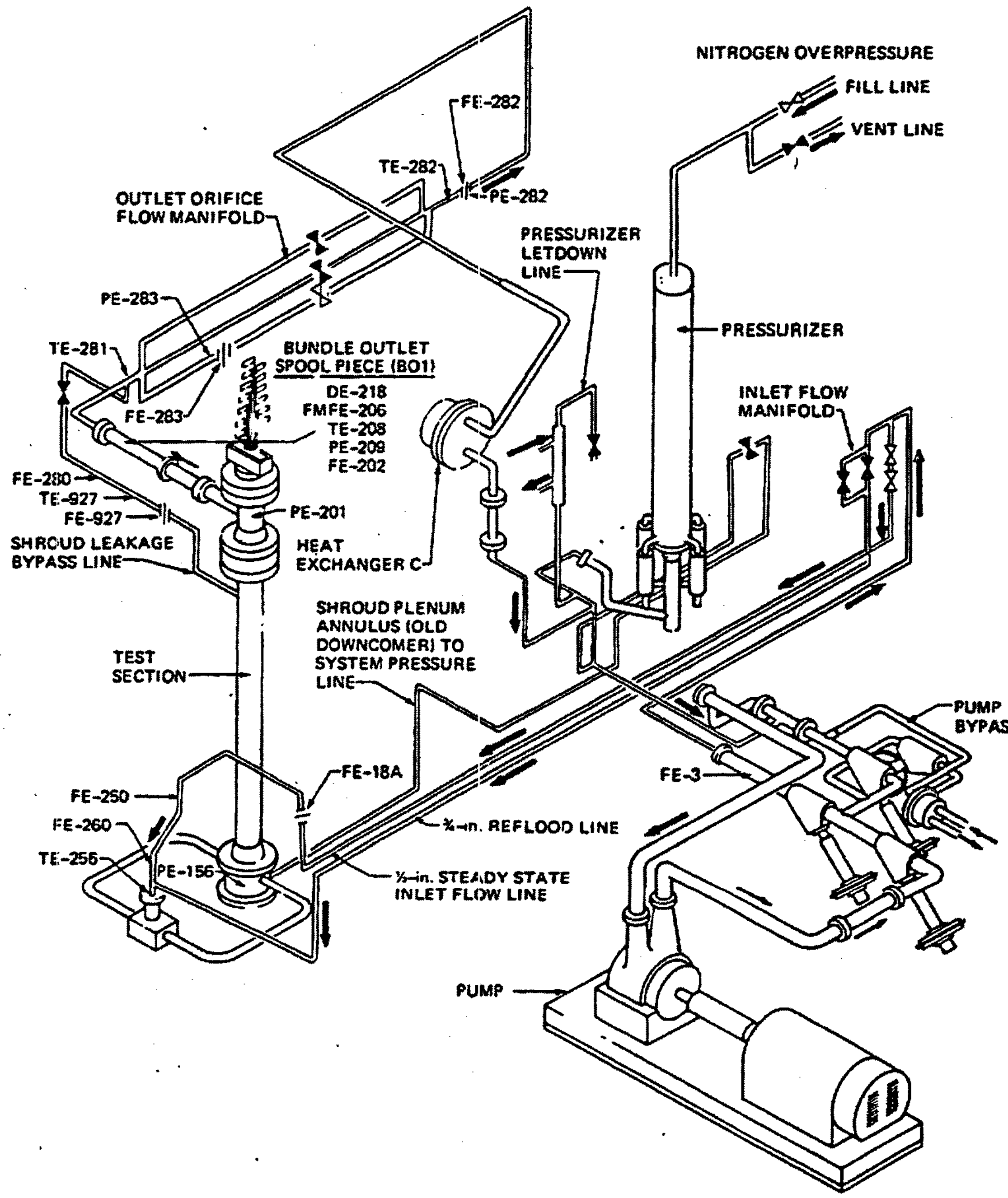


**Figure 2: THTF In-Bundle Pressure Instrumentation (metric units).
(Taken from Anklam et al, 1982.)**

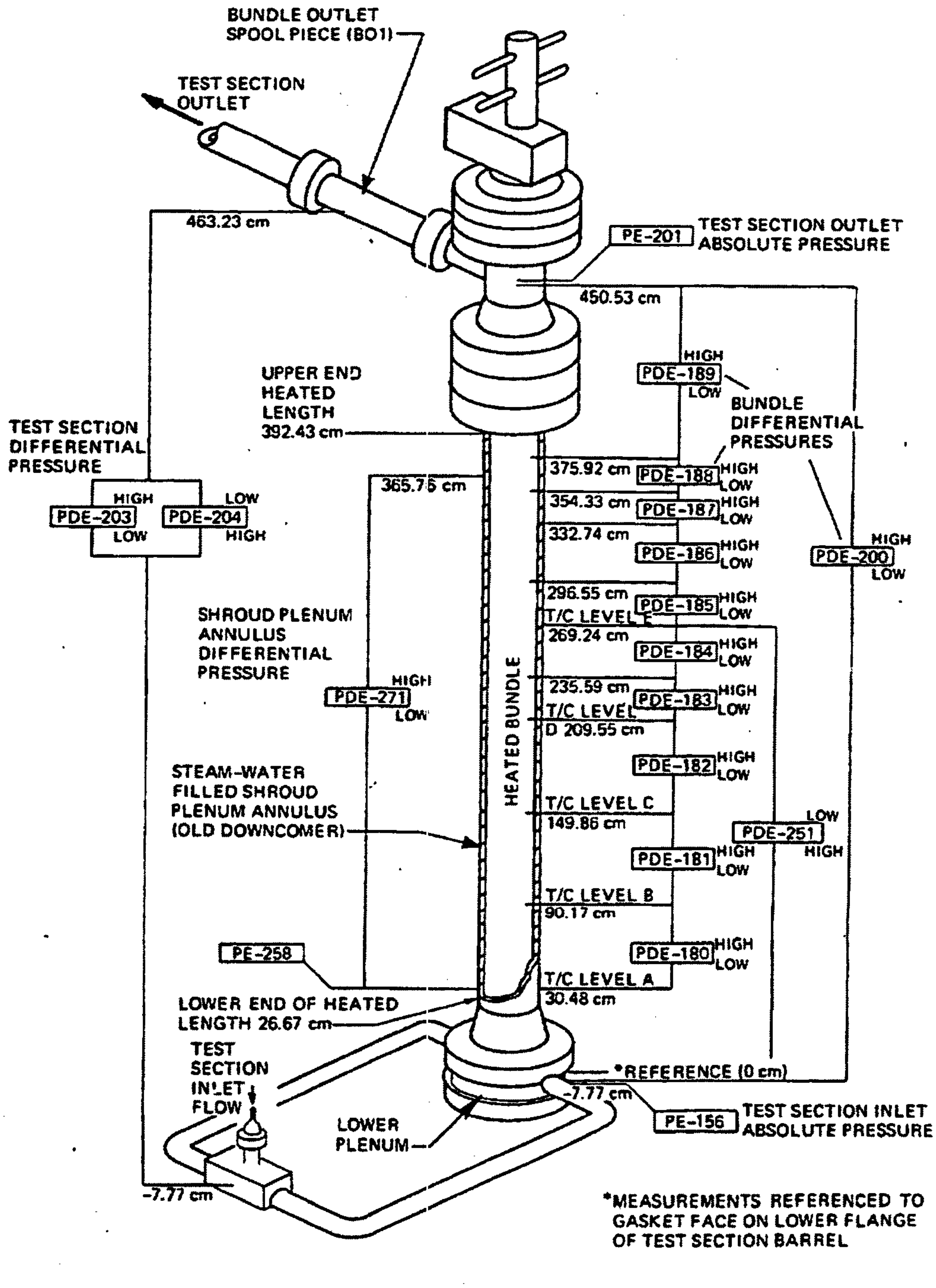


Figure 3: Cross Section of THTF Test Section.
(Taken from Anklam et al., 1982.)

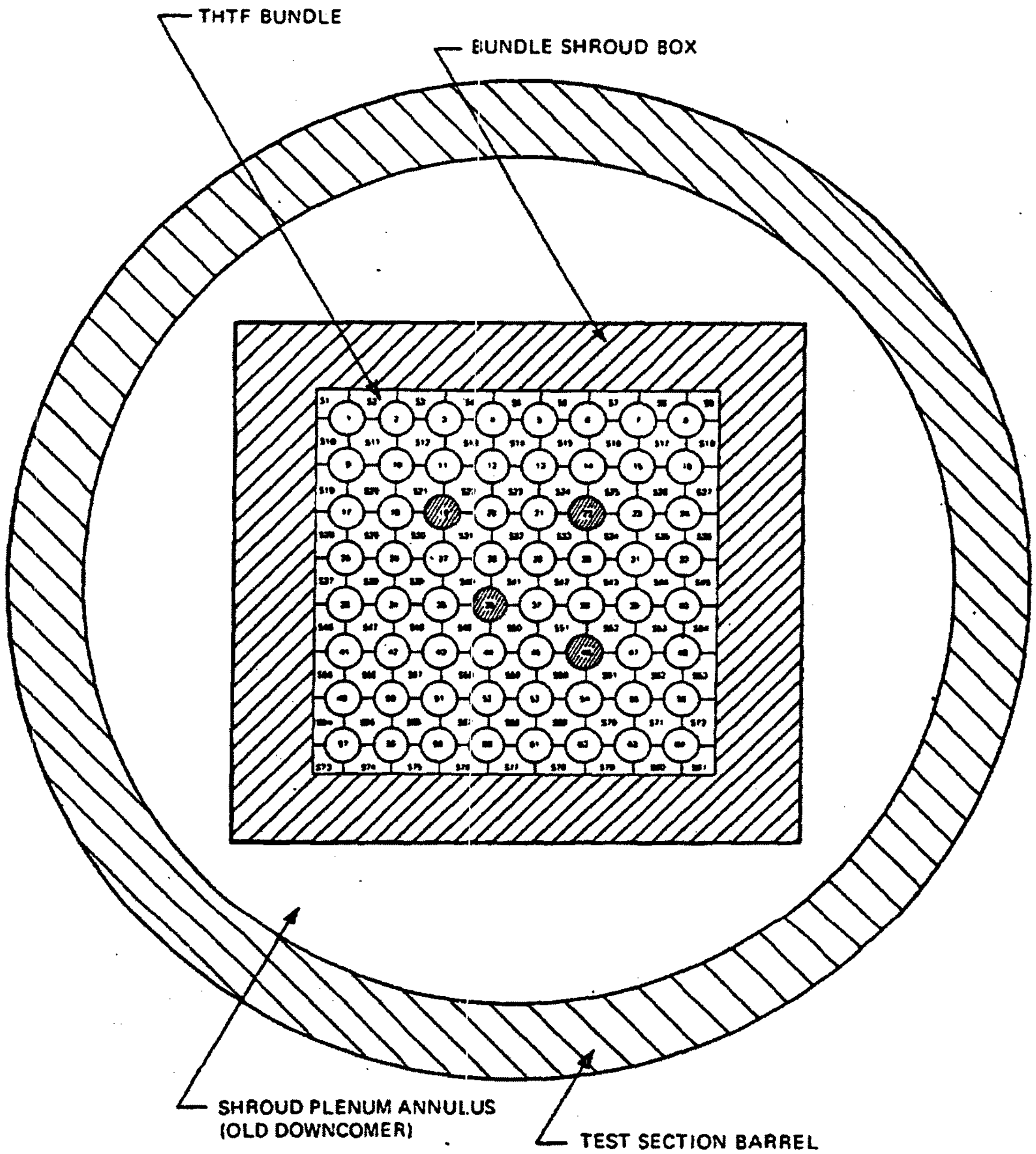


Figure 4: Cross Section of THTF Bundle 3.
(Taken from Anklam et al., 1982.)

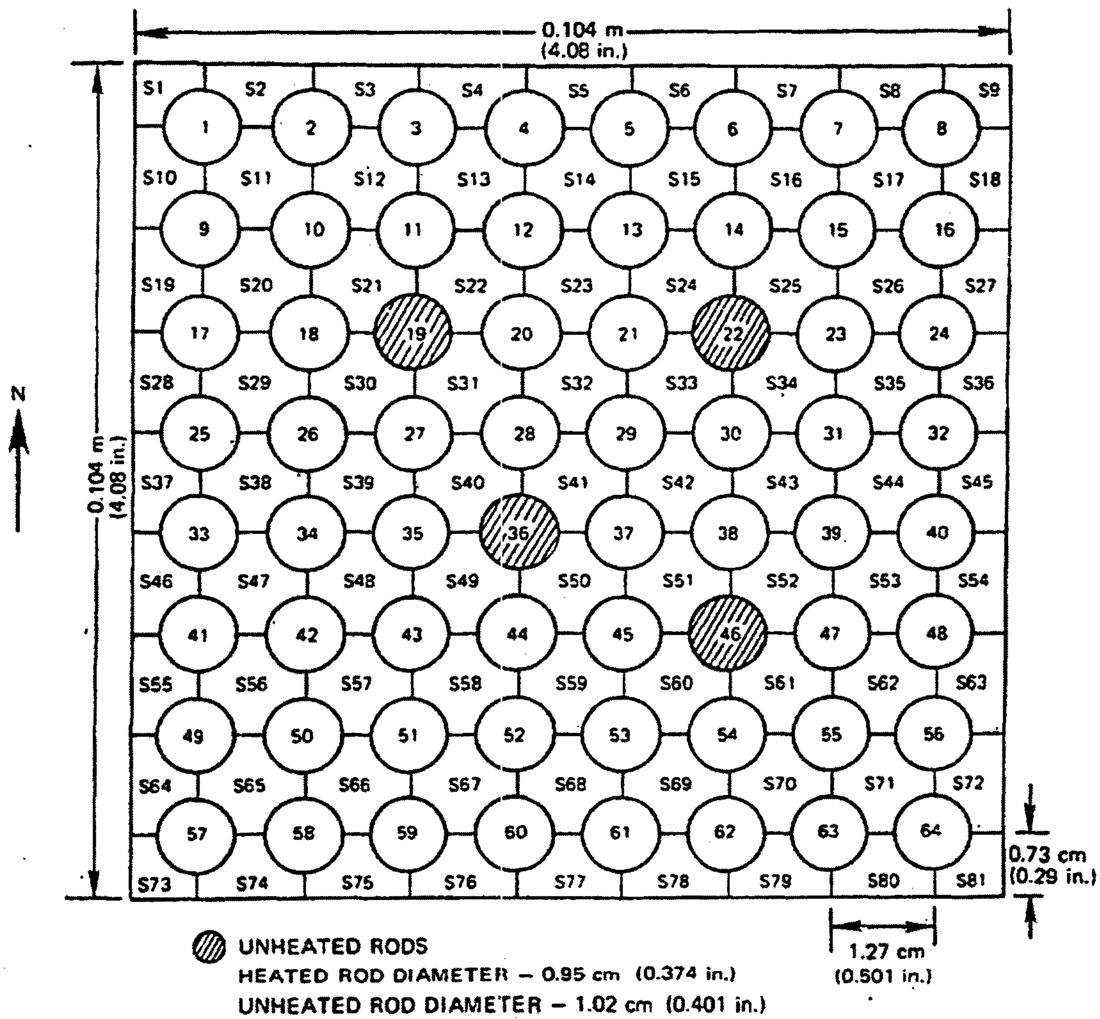


Figure 5: Axial Location of Spacer Grids in Fuel Rod Simulator Thermocouples. (Taken from Anklam et al., 1982.)

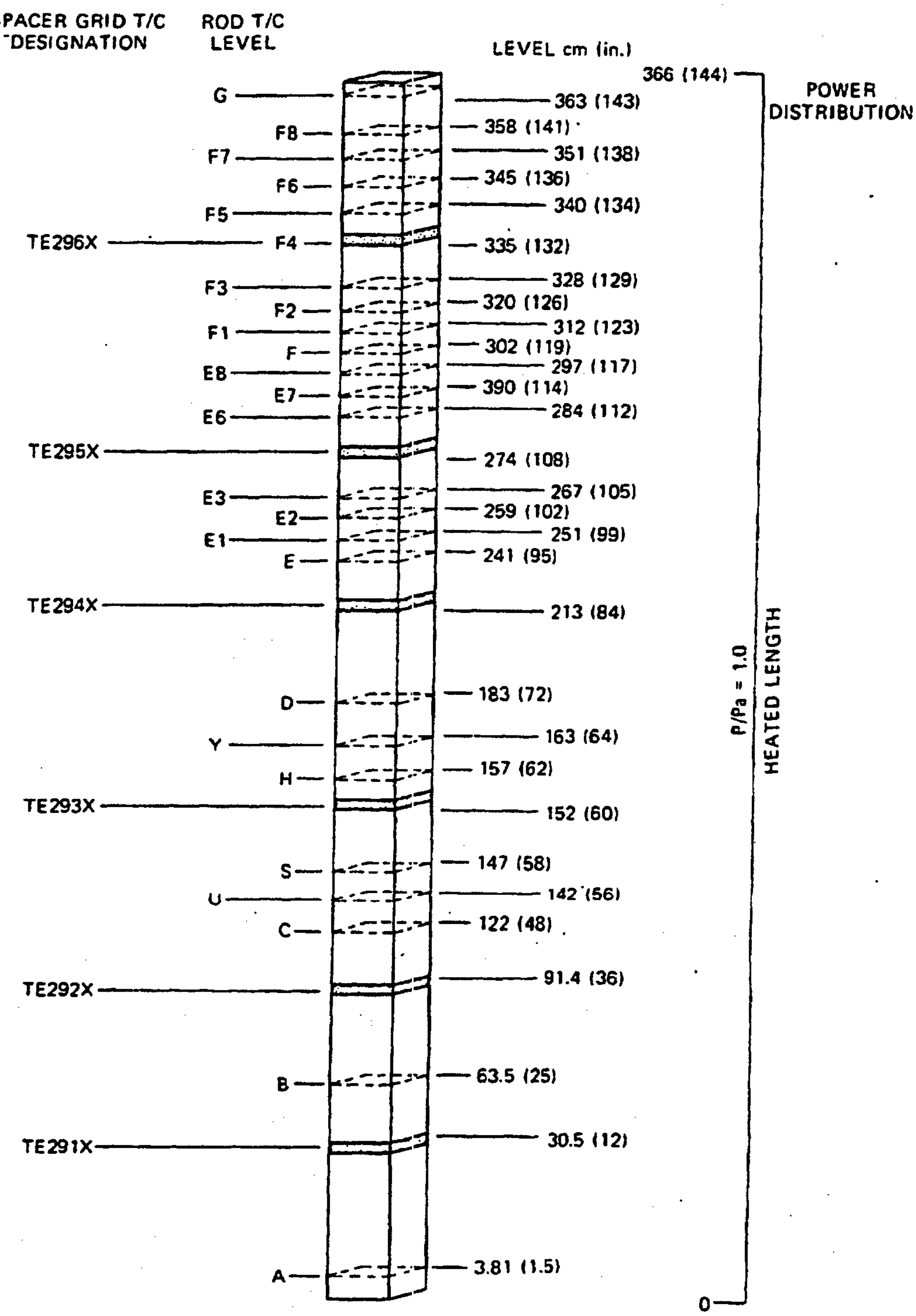


Table 1: Mixture Level Swell Test Matrix[a] (Taken from Anklam et al, 1982)

Test	System Pressure [MPa (psia)]	Linear Power/Rod [kW/m (kW/ft)]	Vapor[b] Superficial Velocity at Mixture Level [m/s (ft/s)]	Mixture Level [m (ft)]	Collapsed[c] Liquid Level [m (ft)]	Average Void Fraction
3.09.10I	▲ 4.50 (650)	2.22 (0.68)	1.30 ± 0.04 (4.25 ± 0.13)	2.62 ± 0.04 (8.60 ± 0.13)	1.34 ± 0.03 (4.39 ± 0.1)	0.49
3.09.10J	▲ 4.20 (610)	1.07 (0.33)	0.61 ± 0.02 (1.99 ± 0.07)	2.47 ± 0.04 (8.10 ± 0.14)	1.62 ± 0.03 (5.31 ± 0.1)	0.34
3.09.10K	▲ 4.01 (580)	0.32 (0.10)	0.15 ± 0.02 (0.50 ± 0.05)	2.13 ± 0.30 (6.98 ± 0.98)	1.62 ± 0.03 (5.31 ± 0.1)	0.24
3.09.10L	● 7.52 (1090)	2.17 (0.66)	0.73 ± 0.02 (2.39 ± 0.06)	2.75 ± 0.09 (9.02 ± 0.29)	1.76 ± 0.03 (5.77 ± 0.1)	0.36
3.09.10M	● 6.96 (1010)	1.02 (0.31)	0.37 ± 0.01 (1.20 ± 0.03)	2.62 ± 0.04 (8.60 ± 0.13)	1.89 ± 0.03 (6.20 ± 0.1)	0.28
3.09.10N	● 7.08 (1030)	0.47 (0.14)	0.12 ± 0.01 (0.40 ± 0.04)	2.13 ± 0.03 (6.98 ± 0.98)	1.86 ± 0.03 (6.10 ± 0.1)	0.13
3.09.10AA	▲ 4.04 (590)	1.27 (0.39)	1.04 ± 0.03 (3.40 ± 0.10)	3.42 ± 0.03 (11.23 ± 0.09)	2.00 ± 0.03 (6.56 ± 0.1)	0.42
3.09.10BB	▲ 3.86 (560)	0.64 (0.20)	0.48 ± 0.02 (1.59 ± 0.07)	3.31 ± 0.04 (10.85 ± 0.12)	2.32 ± 0.03 (7.61 ± 0.1)	0.30
3.09.10CC	▲ 3.59 (520)	0.33 (0.10)	0.40 ± 0.02 (1.31 ± 0.07)	3.60 ± 0.02 (11.80 ± 0.08)	2.88 ± 0.03 (9.45 ± 0.1)	0.20
3.09.10DD	● 8.09 (1170)	1.29 (0.39)	0.46 ± 0.01 (1.50 ± 0.03)	3.23 ± 0.04 (10.61 ± 0.13)	2.39 ± 0.03 (7.84 ± 0.1)	0.26
3.09.10EE	● 7.71 (1120)	0.64 (0.19)	0.27 ± 0.01 (0.88 ± 0.03)	3.47 ± 0.03 (11.40 ± 0.08)	2.85 ± 0.03 (9.35 ± 0.1)	0.18
3.09.10FF	● 7.53 (1090)	0.32 (0.98)	0.12 ± 0.01 (0.40 ± 0.03)	3.23 ± 0.04 (10.61 ± 0.13)	2.90 ± 0.03 (9.51 ± 0.1)	0.10

[a] Some rounding off of numbers has been done. Accordingly, conversions between metric and English and the value of mixture level swell may not appear to be exact.
[b] Defined as the total core volumetric vapor generation rate/unit flow area.
[c] Hydrostatic head of test section liquid inventory.

recorded for each test. For saturated water entering the bottom of the reactor core the average void fraction ($\overline{\alpha}$) is given by:

$$\overline{\alpha} = 1 - \frac{\ell_c}{\ell_{2\phi}} \tag{1}$$

where:

- λ_c is the collapsed level and
- $\lambda_{2\phi}$ is the boiled-up level.

Note also that the steam superficial velocity at the top of the core is listed for each test. The comparisons of these average void fractions and the predictions from function VFVOL are given in the next section.

2.2 FLHT Test Facility

This in-reactor test apparatus is also a full length facility as shown in Figure 6. However, the flow cross-section (see Figure 7) is considerably smaller (12 pins compared to 64) than that used in the THTF experiments. The role of heated and unheated surfaces including the differences between these two experiments are discussed in the next section.

FLHT Experiment 4 measured the collapsed water level using a Time Domain Reflectometer (TDR) as well as the onset of the fuel pin clad overheating (see Figure 8). Table 2 lists the conditions used for Test 4 and it is important to note that the inlet water temperature is highly subcooled. The effective subcooled length needs to be calculated and subtracted from the total mixture swell level when comparing the experimental data with the prediction of function VFVOL. This is discussed in Section 3.

2.3 IIST Facility

The IIST facility is a reduced height, reduced pressure scaled facility for a 3-loop PWR with inverted U-tube steam generators as illustrated in Figure 9. This test facility has been used to examine a range of accident sequences and in the experiment investigating a small break LOCA (Lee et al., 1989), the test facility also recorded both the collapsed water level as well as the onset of the temperature escalation of the electrical heaters simulating the nuclear fuel pins. This temperature measurement is at the top of the heated length and therefore provides an indication of when the core is uncovered.

Figure 10 shows a schematic of the heater rods in the simulated core which has a cross-sectional flow area of 0.0507 m^2. In the small break LOCA experiment, the heater power was 126 kW and the Reactor Coolant System (RCS) pressure was at approximately 3 bars at the time that the boiled up water level dropped below the top of the core region. Figure 11a shows the collapsed water level while Figure 11b shows the heater sheath temperature measurement which clearly shows

Figure 6: FLHT-4 Test Layout. Levels are Elevations in Inches from the Bottom of the Fuel Column. (Taken from Lanning et al., 1983.)

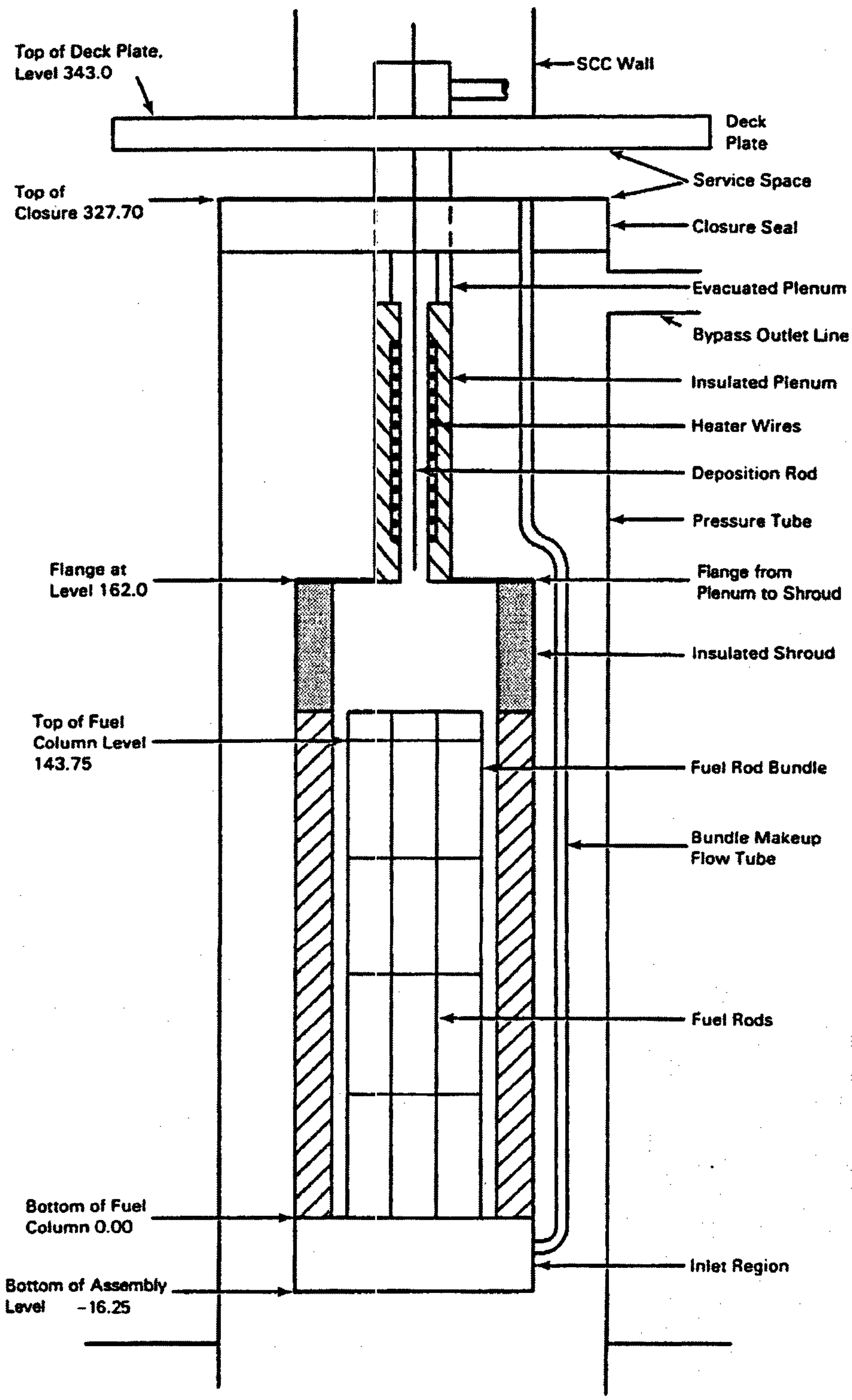
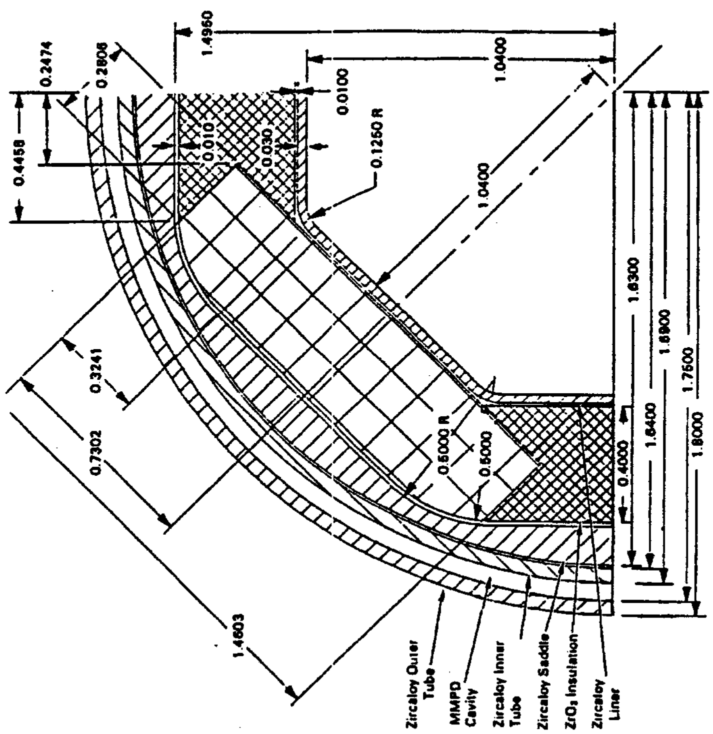
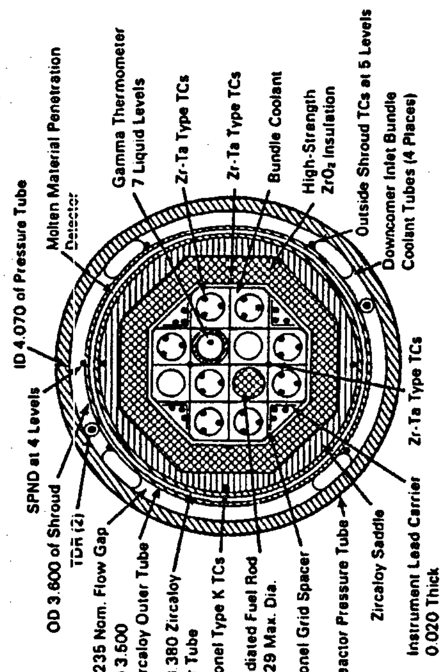


Figure 7: Cross Section of FLHT-4 Fuel Rod Bundle and Shroud:
(a) Full Cross Section; (b) Detail of the FLHT-4 Shroud (dimensions in inches).
(Taken from Lanning et al., 1983.)



(b)



(a)

Figure 8: Liquid Level Versus Time.
(Taken from Lanning et al., 1983.)

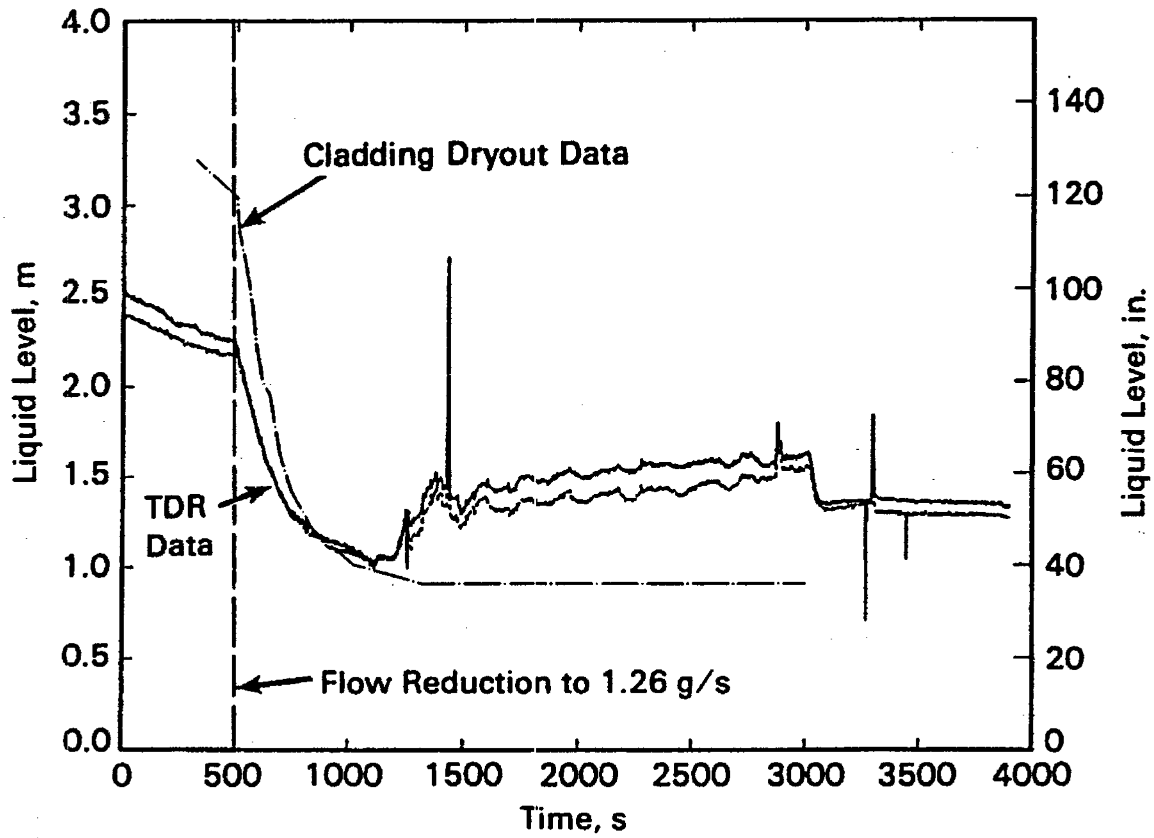


Table 2:

FLHT-4 Boilaway Operating Conditions

Test Section	Measurement
Coolant	Makeup water
Inlet Temperature	358 K
Makeup flow rate	1.26 ± 0.1 g/s
Steam outlet temperature	< 644 K
Outlet pressure	1.4 MPa
Peak cladding temperature	> 2600 K
Total fission power	23 kW
Average linear rod power	0.6 kW/m
Maximum linear rod power	1.07 kW/m
Axial peaking factor	1.5
Radial peaking factor	1.0

Figure 9: Schematic of the IIST Facility (taken from Lee et al., 1989).

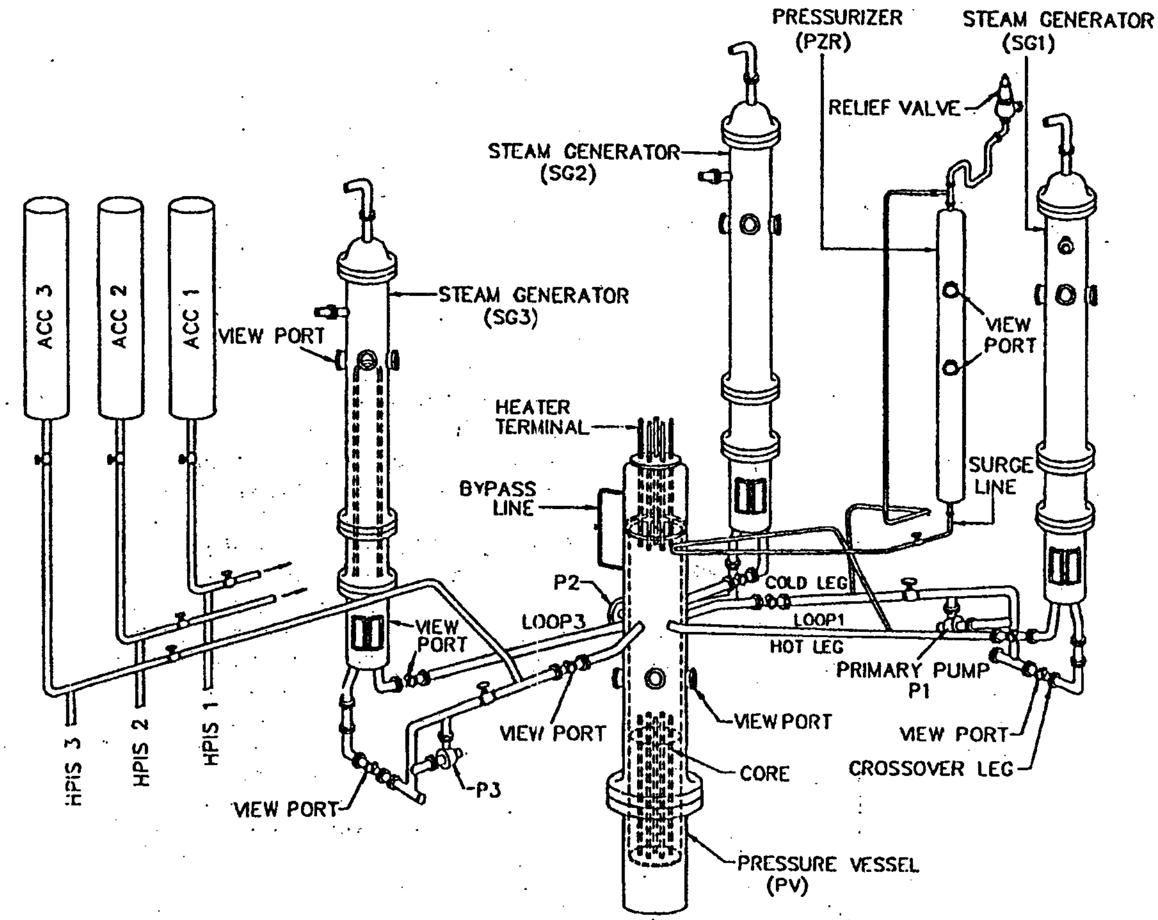


Figure 10: Schematic of the IIST Core Heater Rods Layout.

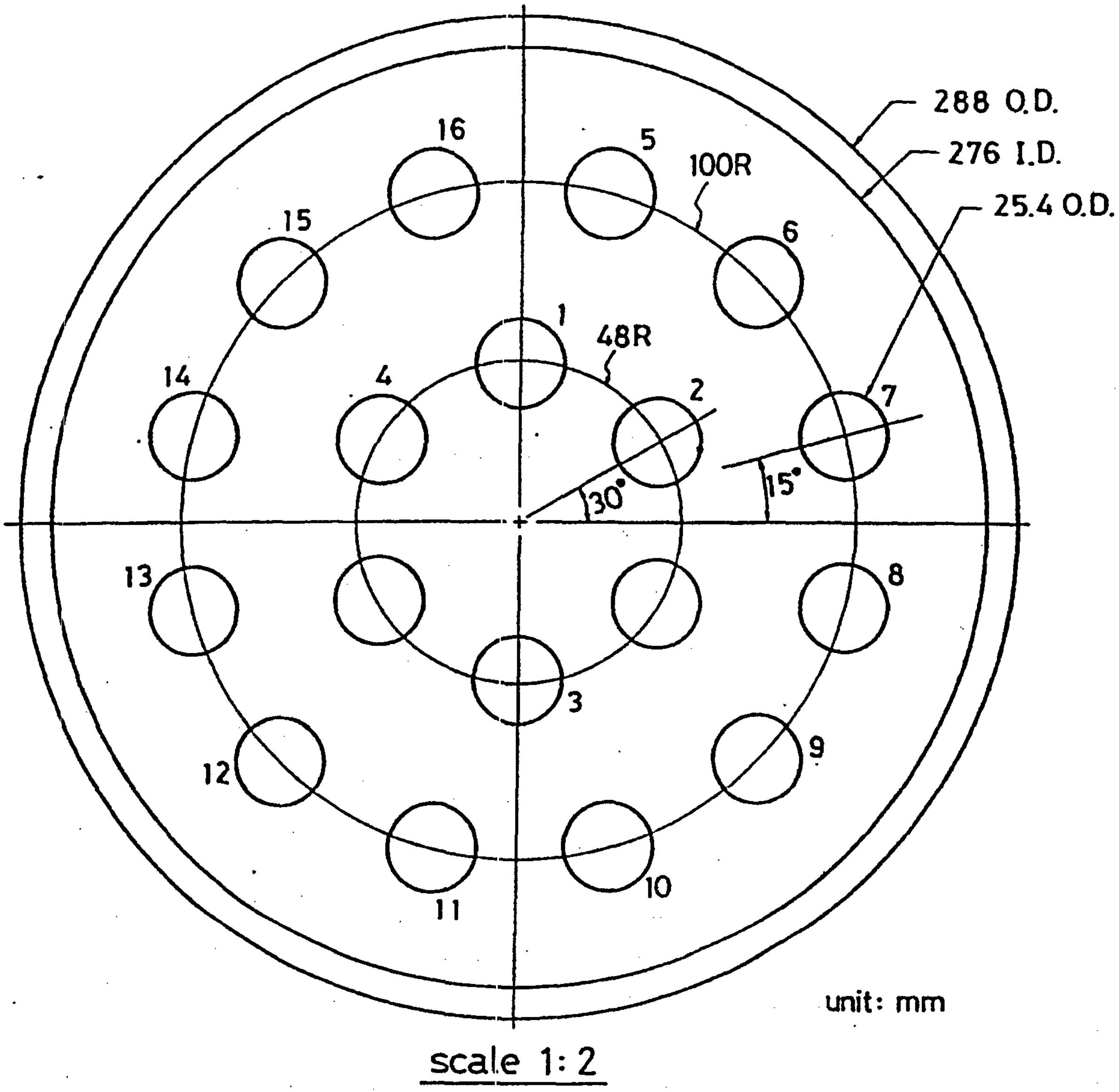


Figure 11a: Pressure Vessel Collapsed Liquid Level (taken from Lee et al., 1989).

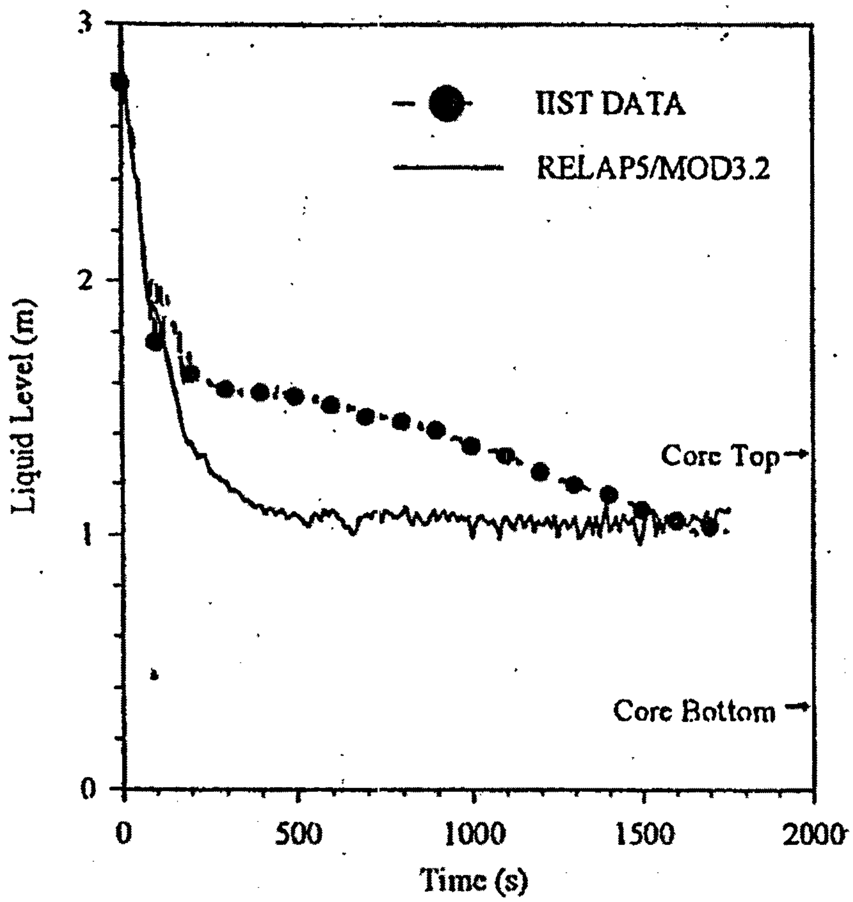
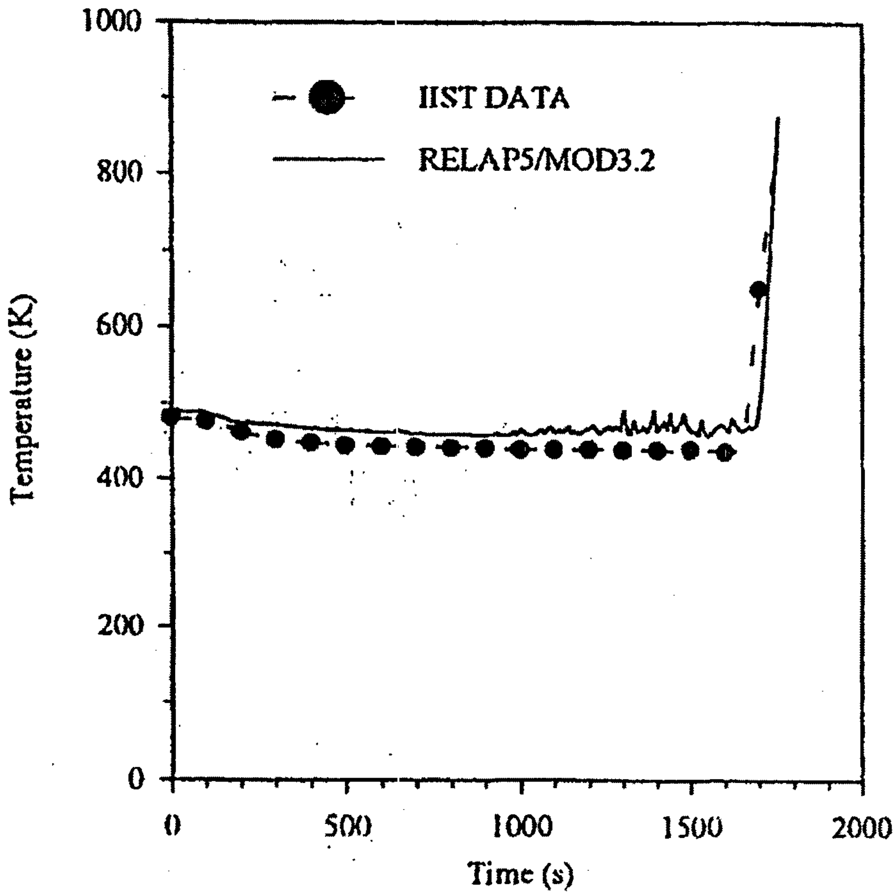


Figure 11b: Core Cladding Temperature (taken from Lee et al., 1989).



the time of overheating for the top of the heated region. From this information, we can deduce average void fraction in the core region when this overheating begins.

3.0 DISCUSSION OF RESULTS

Function VFVOL assesses the average void fraction in a volumetrically heated boiling pool based upon the drift flux relationship:

$$\bar{\alpha} = \frac{\phi}{2 + C_o \phi} \tag{2}$$

where:

$$\phi = \frac{j_g}{U_\infty} \tag{3}$$

and

$$U_\infty = 1.53 \left(\frac{\sigma g \left(\rho_w - \rho_g \right)}{\rho_w^2} \right)^{0.25} \tag{4}$$

In the above expressions, the variables are defined by:

- C_o equals the drift flux proportionality constant that is considered to have uncertainty boundaries between 1.0 and 2.5, with 2.0 being the nominal value.
- j_g is the gas/vapor superficial velocity at the top of the boiling pool,
- U_∞ is the churn turbulent bubble rise velocity,
- g is the gravitational acceleration,
- σ is the liquid-gas surface tension,
- ρ_g is the gas/vapor density, and
- ρ_w is the water density.

As can be seen from the above equations, since it is usually the case that $\rho_w >> \rho_g$, the churn turbulent superficial velocity is generally well represented by

$$U_\infty = 1.53 \left[\frac{g\sigma}{\rho_w} \right]^{0.25} \tag{5}$$

This velocity is only weakly dependent upon the system pressure and is typically about 0.23 m/sec. Therefore, one only needs to know the superficial vapor velocity at the top of the boiling pool and the drift flux proportionality constant to assess the average void fraction. In the evaluations for the different experiments, the nominal distribution parameter is taken to be 2 and the influence of the uncertainty boundaries are also investigated.

3.1 THTF Tests

The information taken from the THTF pool boil-up experiments is listed in Table 1, and in general, can be directly compared to the calculations from function VFVOL. This assumes that the inlet temperature for the individual tests is the saturation temperature corresponding to the system pressure since inlet subcooling does need to be included in the evaluation. The manner for accounting for subcooling is to simply assume that the enthalpy of the coolant passing through the subcooled and two-phase regions experiences an increase given by

$$\Delta h_L = h_g - h_{w,sub} \tag{6}$$

where:

- h_g is the enthalpy of saturated steam at the system pressure, and
- $h_{w,sub}$ is the enthalpy of the subcooled water entering the heated region.

The effective pool length that is occupied in heating up the subcooled water to saturation is given by

$$\Delta h_{sub} = h_f - h_{w,sub} \tag{7}$$

where:

- h_f is the enthalpy of saturated water at the system pressure.

The length of the heated zone that is essentially subcooled water is then given by

$$y = \left[\frac{\Delta h_{sub}}{\Delta h_L}\right] \square \ell_{2\phi} \tag{8}$$

Given an inlet subcooling, the average void fraction is then evaluated by

$$\overline{\alpha} = \frac{\ell_{2\phi} - \ell_c}{\ell_{2\phi} - y} = \frac{1 - \dfrac{\ell_c}{\ell_{2\phi}}}{1 - \dfrac{\Delta h_{sub}}{\Delta h_L}} \tag{9}$$

As indicated by this expression, the influence of inlet subcooling is to increase the average void fraction over that which would be calculated by strictly using the boiled up height and collapsed heights relative to the bottom of the heated length.

For the THTF tests, it is assumed that the inlet subcooling is negligible. Consequently, the data given in Table 1 can be compared directly with the predictions of function VFVOL. These are given in Figure 12 for two different system pressures, i.e. 4 and 7 MPa. Two observations are apparent in this illustration:

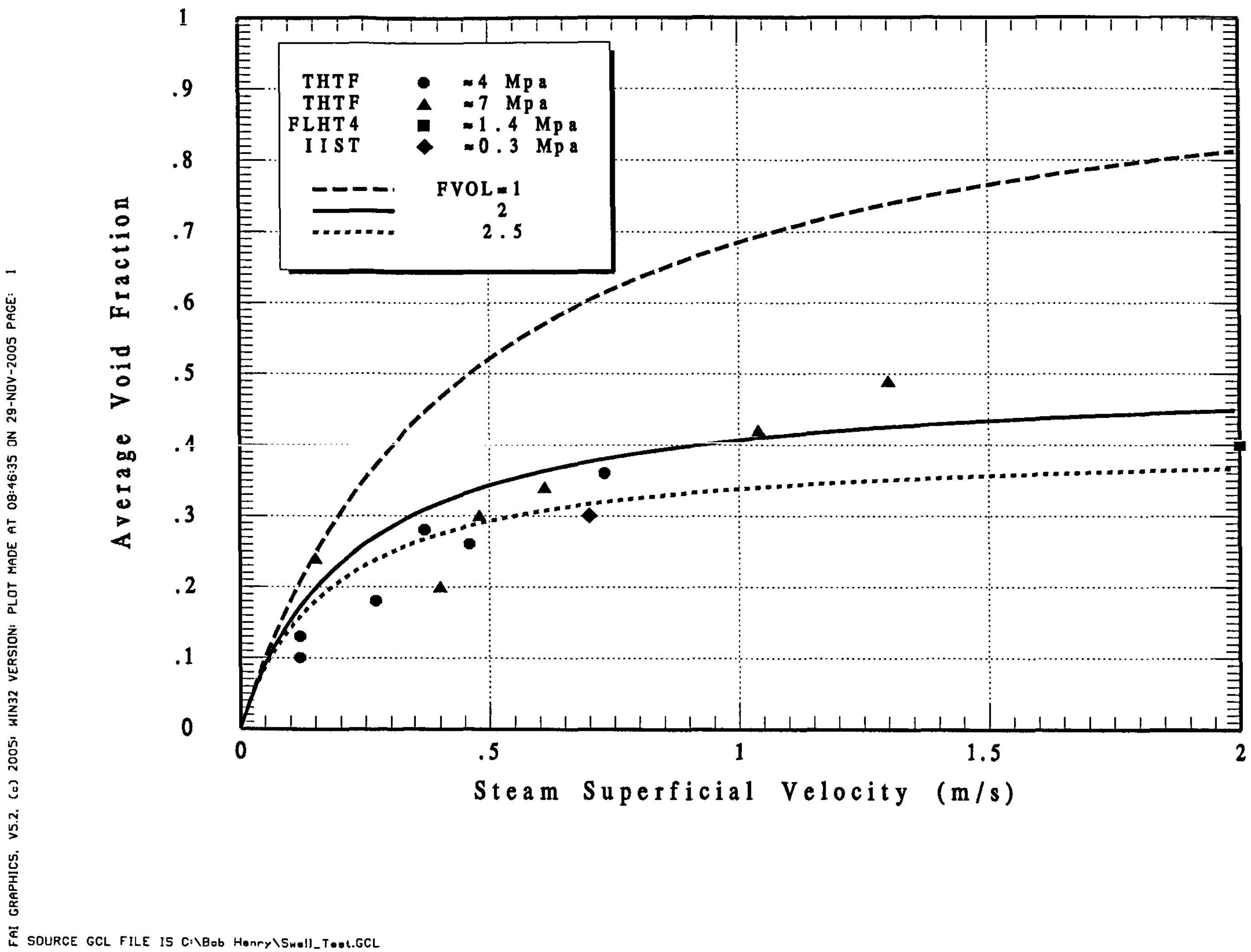


Figure 12: Comparison of the VFVOL Drift Flux Model and FLHT and IIST Experiments.

1. The nominal value of the drift flux parameter C_o provide a good representation of the THTF level swell data.

2. As noted previously, there is no pressure dependence of the information at different pressures when interpreted in this manner.

When evaluating level swell data from limited scale experiments, it is important to consider the ratio of heated to unheated surfaces. Heated surfaces are experiencing steam formation and therefore have the steam-water interactions of interest in the model. Conversely, unheated surfaces do not have this interaction and can be a location that collects water as a film and enable the water to drain. As such, unheated surfaces tend to reduce the boiled-up height and likewise the pool void fraction. Therefore, limited scale experiments with a higher ratio of unheated to heated surfaces could be expected to have somewhat smaller void fraction or level swell compared to the reactor core behavior.

Open lattice PWR cores have very little unheated surface and therefore would be expected to be similar to the level swell in a large pool. BWR cores have fuel assemblies and cans with each can having an eight-by-eight fuel pins arrangement. Hence, a BWR fuel assembly would be expected to behave the same as the THTF test facility with respect to the level swell. PWR core configurations would have a similar or somewhat greater level swell. However, the FLHT test facility with a smaller number of fuel pins and a larger ratio of unheated-to-heated surface would be expected to have a reduced level swell (and void fraction) compared to the THTF test results.

3.2 FLHT Tests

Figure 8 shows the results for test #4 with the boiled-up level being about 3.1 m with the collapsed level at 2.25 m. Table 2 also lists the inlet water temperature at 358 K (85°C) with a system pressure of 1.4 MPa with a saturation temperature of about 473 K (200°C). Therefore, the heated length required to bring the incoming cold water to saturation is about 0.73 m. Substituting this into Equation 9 gives a value for the average void fraction of 0.4 in the saturated region. This is also in agreement with the prediction of function VFVOL for the higher superficial steam velocity of about 2 m/sec. Hence, even with the nonprototypic unheated surface area, the level swell is in agreement with the MAAP4 model as shown in Figure 12.

3.3 IIST SBLOCA Test

Reviewing Figure 11a shows that the core is one meter high and that the collapsed water level at the time that overheating occurs is approximately 0.7 m. Consequently, the average void fraction in the core region at the time that overheating occurs is 0.30%. With the power level, cross-sectional flow area and system pressure mentioned earlier, and assuming that the incoming water is saturated, the superficial steam velocity through the heated bundle is 0.69 m/sec. As illustrated in Figure 12, this point is in very good agreement with the results from the full height, full pressure THTF tests which has a similar superficial steam velocity. It is also important to note that this small scale system, has a much larger ratio of unheated to heated surface within the test bundle. As noted earlier, this tends to skew the results towards a lower average void fraction within the heated bundle. Hence, while this lies on the lower boundary considered for the uncertainty in boiled up level calculations, a large core configuration would be expected to have a higher average void fraction than is demonstrated by these scaled tests. Nonetheless, they provide a good representation of the lower bound for the boiled up pool behavior.

6.0 REFERENCES

Anklam, T. M., et al., 1982, "ORNL Rod Bundle Heat Transfer Test Data Volume 4 ORNL Small Break LOCA Heat Transfer Test Series II: Experimental Data Report," NUREG/CR-2525, Vol. 4.

Lanning, D. D., et al., 1983, "Data Report: Full-Length High-Temperature Experiment 4," Pacific Northwest Laboratory Report PNL-6368.

Lee, C. H., et al., 1989, "Using an IIST SBLOCA Experiment to Assess RELAP5/MOD3.2," Nuclear Technology, 126, pp. 48-61.

Enclosure 2 of L-06-003

Operator Action Times

This enclosure contains the FENOC response to the additional question relative to the operator action times previously identified in FENOC letter L-05-154 dated October 7, 2005.

Reason for the contained additional information:

During a telephone call held on January 9, 2006, with the NRC reviewers, FENOC was requested to provide additional information regarding the operator action times provided in Enclosure 3 of FENOC letter L-05-154 dated October 7, 2005.

Specifically, the following question was asked:

Question:

In the supplemental information submitted in FENOC letter L-05-154 Enclosure 3, the licensee provided Operator Action Times used in the EPU analysis in Table 3-1 of the enclosure. We are now seeking the actual times it took the operator to perform each of the actions as validated by walk-through or assessed by simulator to confirm that the licensee can complete the actions within the times used in the EPU analysis. We would like specification of whether the times were assessed by walk-through or simulation along with the actual times as well.

Response:

The operators are trained on emergency and abnormal operating procedures, and as part of the procedural change process, all of the crews require training on the changes as a result of EPU. The operators' ability to perform the functions required does not change for EPU.

The BVPS-1 operator action times for EPU conditions have been validated for the revised Emergency Operating Procedures (EOP) and confirmed on the BVPS-1 simulator for control room actions as part of the EOP review process. Actions outside the control room were validated via operator walkdowns as part of this procedural change process. The results confirmed that the actions can be completed within the times used in the BVPS-1 EPU analysis.

The revisions to BVPS-2 EOPs have not been completed at this time, and therefore the final BVPS-2 operator action times will be validated as part of the procedural change process, as noted previously in our October 7, 2005 submittal (L-05-154). However, the BVPS-2 response times for EPU conditions were confirmed by operator talk-throughs or walk-throughs using senior licensed operators (SRO's) who are currently licensed at BVPS-2 and were previously licensed at BVPS-1. Also utilized to confirm the operator action times was an SRO who is currently licensed at BVPS-1 and was previously licensed at BVPS-2. The senior licensed operators have performed similar transient scenarios on the simulator and in the field at both units. Input was received from the Operations Management team on the talk-through times at BVPS-2. This input was valuable from the perspective of validating consistent operator crew performance during the transient scenarios between both units. The results are documented in Tables 2-1 and 2-2 which confirm that the operator actions can be accomplished within the times used in the BVPS-2 EPU analysis.

Tables 2-1 and 2-2 contain the requested information regarding the comparison of the BVPS-1 and BVPS-2 operator action times and the times used in the EPU analysis. The EPU and current analysis times are provided to identify the changes. In summary, the loss of coolant accident (LOCA), non-LOCA and main steamline break (MSLB) operator action times are not significantly impacted by EPU. The operator action times assumed within the steam generator tube ruptures (SGTR) overfill analysis did shorten, but the overall time to safety injection (SI) termination was not impacted and this total time is the critical time for the overfill analysis. The majorities of the action times are the same or have increased for EPU. For the action times that have decreased for EPU, the results confirmed that the actions can be completed within the times used in the BVPS-1 EPU analysis.

Table 2-1
Comparison of BVPS-1 Operator Action Times - EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time (from EOP Validation) [Method]	Action Time Completed in Time Available (YES/NO)
Loss of Coolant Accident (LOCA)					
LOCA Switchover from Cold Leg Recirculation to Hot Leg Recirculation	Initiate switchover to simultaneous hot leg/cold leg recirculation	At the following times after the start of the event: 8.0 hours	At the following times after the start of the event: 6.5 hours	[See Note 2] [Simulator]	YES
Non-Loss of Coolant Accident (Non-LOCA)					
Non-LOCA Uncontrolled Boron Dilution (Modes 1, 2 and 3)	Terminate uncontrolled boron dilution flow to the RCS	Within 15 minutes after the start of the event for all Modes	Within 15 minutes after the start of the event for all Modes	22 seconds [Simulator]	YES
Non-LOCA Main Feedline Break	Terminate auxiliary feedwater flow to the faulted steam generator	Within the following times after low-low level is reached in the faulted SG: 10 minutes	Within 15 minutes after low-low level is reached in the faulted SG	7.8 minutes [Simulator]	YES
Non-LOCA Spurious SI – Pressurizer Overfill	Terminate high head safety injection flow to the RCS	There is no current power pressurizer overfill analysis	Within 10 minutes after the start of the event	9.7 minutes [Simulator]	YES
Non-LOCA Loss of Offsite Power – Pressurizer Overfill (due to charging/letdown malfunction)	Mitigate uncontrolled charging flow to the RCS in conjunction with no letdown flow from the RCS	Within 10 minutes after the start of the event	Within 10 minutes after the start of the event	6.3 minutes [Simulator]	YES

Table 2-1 (Continued)
Comparison of BVPS-1 Operator Action Times - EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time (from EOP Validation) [Method]	Action Time Completed in Time Available (YES/NO)
Steam Generator Tube Rupture (SGTR)					
SGTR Overfill Analysis [See Note 1]	1. Isolate auxiliary feedwater flow to the ruptured SG	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reactor trip: 6.8 minutes	4.1 minutes [Simulator]	YES
	2. Isolate steam flow (close MSIV) from the ruptured SG	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reactor trip: 16.7 minutes	8.1 minutes [Simulator]	YES
	3. Initiate cooldown from the intact SGs via the main steam system after MSIV closure	There is no current power analysis for actions inside the main control room or LOFTTR2 SG overfill operational analysis	Within the following times after the MSIV is closed: 1. For actions from inside the main control room: 2.4 minutes [See note 3] Cooldown initiation within 19.1 minutes of the Reactor trip.	[See Note 3] Cooldown initiation was validated at 11.7 minutes of the Reactor trip. [Simulator]	YES
			2. For actions from outside the main control room: 10.0 minutes	6 minutes [Walkthrough]	YES
	4. Initiate RCS depressuriztion (open pressurizer PORV) after completion of the cooldown	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reaching the end of cooldown target temperature: 3.0 minutes	58 seconds [Simulator]	YES
	5. Terminate SI (isolate the high head safety injection flow path) after completion of RCS depressurization	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reaching the end of RCS depressurization target pressure: 4.9 minutes	1.8 minutes [Simulator]	YES

Table 2-1 (Continued)
Comparison of BVPS-1 Operator Action Times - EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time (from EOP Validation) [Method]	Action Time Completed in Time Available (YES/NO)
Main Steam Line Break (MSLB) Mass and Energy (M&E) Releases					
MSLB M&Es Inside Containment (for containment response)	Isolate auxiliary feedwater flow to faulted SG	Within the following times after the start of the event: 10 minutes	Within 30 minutes after the start of the event	4.6 minutes [Simulator]	YES
MSLB M&Es Outside Containment (basis for M&Es that define EQ profile)	Trip the reactor, isolate main feedwater flow to and steam flow from faulted SG and isolate auxiliary feedwater flow (if applicable) to faulted SG	Within the following times after the start of the event: 10 minutes	Within 30 minutes after the start of the event	25.2 minutes [Simulator]	YES

Notes:

1. The SGTR analysis for BVPS-2 is a licensing basis safety analysis while the SGTR analysis for BVPS-1 is an operational response analysis for operator training purposes. Operator actions are modeled in conjunction with the performance of the RCS and steam generators in the LOFTTR2 computer code to achieve the desired goal of preventing overfill of the ruptured SG.

2. Approved validation scenario provided guidance for the "wait time" (transition time) between procedure ES-1.3, Transfer to Cold Leg Recirculation and the start of ES-1.4, Transfer to Simultaneous Hot Leg and Cold Leg Recirculation. At the completion of ES-1.3 the operator transitions back to procedure and step in effect, E-1. This procedure directs preparing to transition to procedure ES-1.4 after 5.5.hours and transitioning at 6.5 hours. This "wait time" was simulated. Time from initiation of event to completion of ES-1.3 was 30 minutes and 40 seconds. Wait time due to procedural requirement to establish recirculation in accordance with ES-1.4 at 6.5 hours was simulated. Transition time to ES-1.4, completion of ES-1.4 and transition back to procedure and step in effect E-1 was 9 minutes and 3 seconds.

3. The SGTR analysis for BVPS-1 is an operational response analysis for operator training purposes. From Westinghouse evaluation for BVPS-1, it was concluded that the elapsed time between ruptured steam generator isolation and cooldown initiation can be greater than the 2.4 minutes (the time identified in the table) with no adverse impact on the conclusions of the SGTR analysis as long as the cooldown is initiated within 19.1 minutes of the reactor trip. As noted in the table, cooldown initiation was validated at 11.7 minutes of the reactor trip.

Table 2-2
Comparison of BVPS-2 Operator Action Times in EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time [Method]	Action Time Completed in Time Available (YES/NO)
Loss of Coolant Accident (LOCA)					
LOCA Switchover from Cold Leg Recirculation to Hot Leg Recirculation	Initiate switchover to hot leg recirculation	At the following times after the start of the event: 7.0 hours	At the following times after the start of the event: 6.0 hours	[See Note 2] [Talk-through]	YES
LOCA Switchover cycling between Hot Leg Recirculation and Cold Leg Recirculation	Initiate cycling between hot leg recirculation and cold leg recirculation	At the following times after the start of the previous initiation of a recirculation alignment: 11.5 hours	At the following times after the start of the previous initiation of a recirculation alignment: 9.5 hours	[See Note 2] [Talk-through]	YES
Non-Loss of Coolant Accident (Non-LOCA)					
Non-LOCA Uncontrolled Boron Dilution (Modes 1, 2 and 3)	Terminate uncontrolled boron dilution flow to the RCS	Within 15 minutes after the start of the event for all Modes	Within 15 minutes after the start of the event for all Modes	1 minute [Talk-through]	YES
Non-LOCA Main Feedline Break	Terminate auxiliary feedwater flow to the faulted steam generator	Within the following times after low-low level is reached in the faulted SG: 15 minutes	Within 15 minutes after low-low level is reached in the faulted SG	8 minutes [Talk-through]	YES
Non-LOCA Spurious SI – Pressurizer Overfill	Terminate high head safety injection flow to the RCS	Within 10 minutes after the start of the event	Within 10 minutes after the start of the event	9.5 minutes [Talk-through]	YES
Non-LOCA Loss of Offsite Power – Pressurizer Overfill (due to charging/letdown malfunction)	Mitigate uncontrolled charging flow to the RCS in conjunction with no letdown flow from the RCS	Within 10 minutes after the start of the event	Within 10 minutes after the start of the event	6 minutes [Talk-through]	YES

Table 2-1 (Continued)
Comparison of BVPS-2 Operator Action Times in EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time [Method]	Action Time Completed in Time Available (YES/NO)
Steam Generator Tube Rupture (SGTR)					
SGTR Overfill Analysis [See Note 1]	1. Isolate auxiliary feedwater flow to the ruptured SG	Within 9.1 minutes after reactor trip	Within the following times after reactor trip: 5.5 minutes	5 minutes [Talk-through]	YES
	2. Isolate steam flow (close MSIV) from the ruptured SG	Within 9.1 minutes after reactor trip	Within the following times after reactor trip: 15.0 minutes	10 minutes [Talk-through]	YES
	3. Initiate cooldown from the intact SGs via the main steam system after MSIV closure	Within 9 minutes after the MSIV is closed for action from outside main control room	Within the following times after the MSIV is closed: 1. For actions from inside the main control room: 2.0 minutes	2 minutes [Talk-through]	YES
			2. For actions from outside the main control room: 7.0 minutes	6 minutes [Talk-through] & [Walk-through]	YES
	4. Initiate RCS depressuriztion (open pressurizer PORV) after completion of the cooldown	Within 2.5 minutes after reaching the end of cooldown target temperature	Within the following times after reaching the end of cooldown target temperature: 4.0 minutes	1 minute [Talk-through]	YES
	5. Terminate SI (isolate the high head safety injection flow path) after completion of RCS depressurization	Within 1.25 minutes after reaching the end of RCS depressurization target pressure	Within the following times after reaching the end of RCS depressurization target pressure: 3.0 minutes	2 minutes [Talk-through]	YES

Table 2-1 (Continued)
Comparison of BVPS-2 Operator Action Times in EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time [Method]	Action Time Completed in Time Available (YES/NO)
Main Steam Line Break (MSLB) Mass and Energy (M&E) Releases					
MSLB M&Es Inside Containment (for containment response)	Isolate auxiliary feedwater flow to faulted SG	Within the following times after the start of the event: 30 minutes	Within 30 minutes after the start of the event	5 minutes [Talk-through]	YES
MSLB M&Es Outside Containment (basis for M&Es that define EQ profile)	Trip the reactor, isolate main feedwater flow to and steam flow from faulted SG and isolate auxiliary feedwater flow (if applicable) to faulted SG	Within the following times after the start of the event: 30 minutes	Within 30 minutes after the start of the event	25 minutes [Talk-through]	YES

Notes:

1. The SGTR analysis for BVPS-2 is a licensing basis safety analysis while the SGTR analysis for BVPS-1 is an operational response analysis for operator training purposes. Operator actions are modeled in conjunction with the performance of the RCS and steam generators in the LOFTTR2 computer code to achieve the desired goal of preventing overfill of the ruptured SG.

2 These operator action times for LOCA switchover from Cold Leg Recirculation (CLR) to Hot Leg Recirculation (HLR) and commencing cycling between Hot Leg Recirculation and Cold Leg Recirculation are based on "waiting times" established within the procedures. After initial CLR is established, the operator waits for the appropriate time as defined in the procedure to implement the identified procedures. These operator actions are performed well after the initiating event and pose no challenge to the operator executing the required procedural steps. At the completion of ES-1.3, "Transfer to Cold Leg Recirculation" the operator transitions back to the procedure and step in effect, procedure E-1. This procedure directs preparing to transition to procedure ES-1.4, "Transfer to Hot Leg Recirculation" and transitioning at the designated time established in the procedure. This "wait time" is mandated by the procedure. This same concept applies for implementing Hot Leg Recirculation back to Cold Leg Recirculation.

Enclosure 3 of L-06-003

Additional Information Regarding the Control Room Atmospheric Dispersion Factors and Dose Consequence Information

The NRC reviewers requested additional information as a follow-up to the November 29-30, 2005, EPU Dose Assessment Audit regarding Control Room atmospheric dispersion factors and dose consequence information. The information was provided in the Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 1) and amendment by FENOC Letter No. L-05-204 (Reference 2).

Specifically, the following was requested:

1. **Additional information was requested pertaining to the iodine-spike, scenario-specific, control room dose consequences following a Main Steam Line Break (MSLB), and a Steam Generator Tube Rupture (SGTR) at EPU conditions.**

   Response:

   The EPU LAR (Reference 1) as amended by FENOC Letter No. L-05-204, dated December 30, 2005 (Reference 2), reported the control room dose consequences of the iodine-spike scenario that resulted in the higher doses. This level of information was considered sufficient since, for the control room, the acceptance criteria is the same for both iodine-spike scenarios.

   Table 3-1 below provides a summary of the post-EPU, BVPS-1 [with Replacement Steam Generators (RSGs)] and BVPS-2, MSLB and SGTR control room dose consequences for both the pre-accident iodine-spike case (PIS), and the concurrent iodine-spike case (CIS), respectively.

Table 3-1
Control Room Operator Dose following EPU (rem TEDE)

Accident	BVPS-1 (w RSGs)	BVPS-2	Regulatory Limit
Main Steam Line Break	0.5 (PIS) 0.66 (CIS)	0.2 (PIS) 0.6 (CIS)	5.00
Steam Generator Tube Rupture	1.95 (PIS) 0.67 (CIS)	0.32 (PIS) 0.13 (CIS)	5.00

2. **Additional information was requested pertaining to control room atmospheric dispersion factors relative to a) methodology and regulatory guidance utilized (specifically for diffused source calculations), b) input data utilized in the ARCON96 runs for selected release points, and c) potential releases from the personnel hatch following a fuel handling accident in the containment.**

Response:

As noted in Section 5.11.9.2 of Reference 1 (sub-section titled "On-Site Atmospheric Dispersion Factors"), the methodology utilized to develop the atmospheric dispersion factors is discussed in detail in Section 5.3.4.2 of the BVPS-1 & 2 LAR Nos. 300 and 172, submitted to NRC via FENOC letter L-02-069 (Reference 3).

As noted in Section 5.3.4.2 of Reference 3, all input data for the ARCON96 runs were developed in accordance with draft NRC guidance on control room habitability assessments, i.e., Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Habitability Assessments at Nuclear Power Plants," December 2001.

In response to questions regarding methodology utilized for diffused source calculations, Section 5.3.4.2 of Reference 3 indicates that only the containment edge release is considered to be a diffuse source as the release is from the entire containment surface. Diffuse source treatment allows the calculation of initial values of the dispersion coefficients (i.e., sigma y and sigma z). These values are determined by the height and width of the containment divided by a factor of six based on DG 1111 (i.e., sigma y = 6.83m and sigma z = 7.44m). All other releases are conservatively treated as point sources.

In response to questions regarding input data utilized in the ARCON96 runs, Tables 3-2 and 3-3 summarize the release point and receptor configuration information, release mode, and meteorological sensor configuration information used as input to ARCON96 for the following release points, i.e., the BVPS-1 Containment Equipment Hatch, the BVPS-2 Main Steam Line Break and Air Ejector (AEJ), and the BVPS-2 Containment Equipment Hatch. The receptors are the BVPS-1 and BVPS-2 control room air intakes.

In response to the question regarding releases from the personnel hatches, atmospheric dispersion factors were not developed for the BVPS-1 and BVPS-2 Personnel Hatches even though releases could occur via these hatches following a fuel handling accident in the containment. The personnel hatches open into the Cable Vault and Rod Control Area, which has a door to the Auxiliary Building, which is open during refueling. These areas are exhausted to the environment via the Supplemental Leak Collection System (release point is the containment top), and via the auxiliary building ventilation (release point is the ventilation vent). Tables 5.11.9-2A & B of Reference 1 provide the atmospheric dispersion factors associated with the BVPS-1 and BVPS-2 containment top and ventilation vent.

References:

1. FENOC Letter L-04-125, License Amendment Requests 302 and 173, dated October 4, 2004.
2. FENOC letter L-05-204, Additional Information Regarding Dose Consequence Analyses in Support of License Amendment Request Nos. 302 and 173, dated December 30, 2005.
3. FENOC letter L-02-069, License Amendment Requests 300 and 172, dated June 5, 2002.

Table 3-2
ARCON96 Atmospheric Dispersion Factor Inputs
BVPS-1 Release Points

ARCON 96 Parameter	BVPS-1 Containment Equipment Hatch	
	BVPS-1 CR Intake	**BVPS-2 CR Intake**
Meteorological Information:		
Period of Meteorological Data	1990 - 1994	1990 - 1994
Lower Measurement Height (m)	10.7	10.7
Upper Measurement Height (m)	45.7	45.7
Wind Speed Units	m/sec	m/sec
Meteorological Data File Names	arconbv.met	arconbv.met
Source Information		
Release Type	ground	ground
Release Height (m)*	0.15	3.6
Building Area (m^2)	1,600	1,600
Vertical Velocity (m/sec)	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0
Stack Radius (m)	0.0	0.0
Receptor Information:		
Distance to Receptor (m)	95.1	114.9
Intake Height (m)	0.15	3.6
Elevation Difference (m)	0.0	0.0
Direction to Source (deg)	195	202
Default Information:		
Surface Roughness Length (m)	0.20	0.20
Wind Direction Window (degrees)	90	90
Minimum Wind Speed (m/sec)	0.5	0.5
Averaging Sector Width Constant	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0

* Release height conservatively set equal to the receptor height.

Table 3-3
ARCON96 Atmospheric Dispersion Factor Inputs
BVPS-2 Release Points

ARCON96 Parameter	BVPS-2 Main Steam Line Break/AEJ		BVPS-2 Containment Equipment Hatch	
	BVPS-1 CR Intake	BVPS-2 CR Intake	BVPS-1 CR Intake	BVPS-2 CR Intake
Meteorological Information:				
Period of Meteorological Data	1990 - 1994	1990 - 1994	1990 - 1994	1990 - 1994
Lower Measurement Height (m)	10.7	10.7	10.7	10.7
Upper Measurement Height (m)	45.7	45.7	45.7	45.7
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names	arconbv.met	arconbv.met	arconbv.met	arconbv.met
Source Information:				
Release Type	ground	ground	ground	ground
Release Height (m)*	0.15	3.6	0.15	3.6
Building Area (m^2)	0.01	0.01	1,600	1,600
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	72.9	53.7	132.3	109.4
Intake Height (m)	0.15	3.6	0.15	3.6
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (deg)	82	95	41	38
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

* Release height conservatively set equal to the receptor height.