Section 5
RESULTS AND SENSITIVITIES

5.1 INTRODUCTION

This section discusses the results of the SHNPP spent fuel pool (SFP) best estimate probabilistic analysis of the seven step Postulated Sequence admitted as a contention in the SHNPP license amendment proceeding. However, in addition, it is judged vital to the decision-makers to provide a characterization of the uncertainty associated with the Base Case evaluation. Therefore, this section also addresses how the uncertainty should be characterized.

5.2 OVERVIEW OF UNCERTAINTY

The Best Estimate is used for decision making because the use of upper bounds (or lower bounds) may introduce biases into the decision making process that are not properly characterized, i.e., the biases may be unevenly applied (widely varying levels of conservatism) with the resulting upper bound yielding a distortion of the importance of individual components of the analysis and potentially of the overall results. Such biases could then lead to improper decisions regarding the importance of individual elements of the analysis. It may also lead to the improper allocation of resources to address conditions or postulated events that have been "conservatively" treated in an upper bound evaluation. Therefore, all prudent evaluations have been included to achieve the Best Estimate characterization.

This Best Estimate analysis is provided in the enclosed evaluation. It is noted, however, that there remain inherent conservatisms in the deterministic calculations, the models, and the assumptions. These "conservatisms" are not able to be extricated from the analysis because the current state of technology is not sufficient to remove them. For example, the assumption that the probability of an exothermic reaction in the SFP is 1.0 is considered to be a default estimate, recognizing both the current state of the technology for calculating the probability of such an SFP exothermic reaction and the low probabilities of the six steps leading to uncovering the spent fuel in SFPs C and D. In light of the information provided by CP&L relating to the "age" of the spent fuel after discharge from the reactor that is to be stored in SFPs C and D, the assumption that an SFP exothermic reaction will occur with a probability of 1.0 is judged to be a conservative assumption. CP&L has addressed qualitatively how unlikely such an exothermic oxidation reaction would be in SFPs C and D. (See Affidavit of Robert K. Kunita.)

The NRC, its contractors, and the industry have committed substantial efforts to the understanding of uncertainties in nuclear power plant risk analyses. These efforts have led to methods development, understanding of the contributors to the uncertainty distributions, and the identification of alternative ways to provide decision makers with effective ways of characterizing the risk spectrum.

There are several sources of uncertainty and several viable ways of categorizing these sources. A simple three category approach is used here [4-22, 4-23]. Each category is then further developed to illustrate more specifically those sources of uncertainty assigned to each category.

The three types or categories of uncertainties are generally considered to be the following:

- Quantification: The related contributors to the so-called "quantification" uncertainties include the following:
  - Failure rate models
  - Applicability of data
  - Statistical variation of parameters
  - Processing simplifications or truncations

- Logic Modeling: The related contributions to logic modeling uncertainties include the following:

    - Adequacy of details
        - Hardware, including instrumentation
        - Human interaction
        - Environmental/spatial
        - Equipment wear out
        - Applicability of data

    - Logic correctness
        - Success criteria
        - Event sequences
        - Systems analysis

    - Dependencies (initiating events, intercomponent, intersystem, functional, environmental, human, and physical similarity)

    Analysis of this category of uncertainties evaluates whether, given the scope of the evaluation, the implementation resulted in models capable of supporting the results, conclusions, and expected use in the support of decisions.

- Scope and Completeness: The considerations include the following:

    - Initial plant conditions (e.g., configurations)
    - End states
    - Inter-unit connections
    - Initiating events
    - Success criteria
    - Event sequence
    - Systems analysis
    - Failure modes and causes
    - Human interaction and errors of commission
    - Data
    - Design deficiencies

Analysis of this category of uncertainties evaluates whether the specific scope is sufficient to support the types of conclusions and decisions reached, and how scope limitations affect the results, conclusions, and decisions that can be supported.

Folded into each of the categories are a set of attributes. These attributes can affect the evaluation of the uncertainty and include the following:

- Plant-Specific: .

  Plants vary in hardware, personnel, procedures, organizations, management, training, etc. These major factors modify the uncertainty associated with accident sequences in each category.

- Time-Varying:

  A specific plant's characteristics will change as a function of plant life due to changes in plant hardware, training, procedures, management, equipment degradation, and aging.

- Sequence-Specific:

  Each accident sequence has unique characteristics that can profoundly affect the ability to quantify the likelihood of such sequences. The sequences vary in the complexity of operator actions, the specific hardware failures, etc.

There are several principles regarding the treatment of uncertainties in probabilistic analyses which have some consensus in the industry. They are identified here to provide a foundation for the scope of this uncertainty evaluation. These principles are as follows:

- The purpose of the uncertainty evaluation is to focus attention on important assumptions.

- Establishing a risk framework for the discussion of point estimate values and their uncertainties provides decision makers additional input.

- The uncertainty process should be usable as an engineering tool to enhance the confidence in the conclusions.

- Attempts to provide a quantitative perspective on uncertainty that is very costly and does not fully support the real objectives of establishing the validity of the conclusions of the assessment or application should be avoided.

- A reasonable, credible range in which the actual value will be found (90 percent degree of belief) is a desirable quantitative measure.

- A Probabilistic Safety Assessment (PSA) process is an engineering applications tool. Therefore, the uncertainty evaluation should be structured in a similar fashion to take maximum advantage of the available engineering insights and to add to those insights. The structure of the approach need not be a rigid formalism, but can, rather, borrow its justification from other published discussions such as the use of a subjectivist approach in risk assessment.

The conclusion from this overview is that the use of focused sensitivity evaluations to characterize the change in the results as a function of changes in the inputs provides a physically meaningful method of conveying the degree of uncertainty associated with the analysis. Therefore, sensitivity cases were developed that portray the changes in the Postulated Scenario frequency as posed by the ASLB, if input variations occur.

The key variations in the 21 sensitivity cases examined address the three categories of uncertainties cited above and adhere to the principles of an effective uncertainty evaluation:

- <u>Quantification</u>: Vary the input accident sequence frequencies and system configuration – See Cases A.1, A..2, A.4, and seismic cases 5.1 through 5.10.

- <u>Logic Modeling</u>: Vary success criteria, human interaction effectiveness, environmental factors, system reliability and dependency effects – See Cases A.3, B.1, B.2, B.3, B.4, B.5, and B.6.

- <u>Completeness</u>: Vary phenomenological effects – See Case C.1.

5.3 SENSITIVITY CASE

The measure of risk used in these analyses is the frequency of the Postulated Scenario (steps 1 through 6). All tables in this section use this parameter to characterize the risk.

The best estimate of the frequency of the loss of effective cooling to the spent fuel has been constructed within the current state of the technology. There are some assumptions that have been included in the model construction and quantification that may introduce some conservatisms. These have been discussed in Section 2.5 and are summarized in the conclusions, Section 6.

The quantitative results are properly considered in two groups: (1) internal events and (2) external events and shutdown events. For internal events, there is high confidence in the models and the evaluation of the SHNPP SFP response to the Postulated Sequence. Most of the effort focused on assessing the impact of the internal events because they are the most studied and lead to the highest frequency of core damage. The results of the internal events initiated sequences indicate that the loss of effective SFP water cooling occurs at a best estimate frequency of 2.65E-8/yr.

The external events and shutdown events were also evaluated to determine whether these events alter the conclusion determined based on the internal events assessment. It is recognized that the uncertainties associated with these sequences are greater than those in the internal events analyses. Consequently, several conservativisms were incorporated in the modeling, which produced inflated point estimate values. Thus, these results are not entirely a "best estimate" because of the conservatisms found in the existing models and generic studies.

Thus, the calculated best estimate annualized probability of the Postulated Sequence based on the internal events analysis is 2.65E-8. This "best estimate" includes the conservative assumption that the conditional probability of step 7 is 1.0. There are also other conservatisms included in the analysis because of the difficulty of removing embedded conservatisms from existing analyses. For example, the time to recover from the loss of cooling to the spent fuel pools was assumed to be four days, based on the maximum heat load in spent fuel pool A after discharge of fuel during refueling. A best estimate calculation could have integrated the reduction in decay heat load over the length of a normal fuel cycle. However, the probability of the Postulated Sequence was already so low, even with numerous conservatisms, that further analysis to refine the calculation was not justified.

The analysis from Section 4 is summarized in Table 5-1, indicating the probability of the Postulated Sequence from internal, fire-induced, seismic and shutdown events. Although this analysis concluded that the best estimate of the probability of the Postulated Sequence is represented by the contribution of internal events only, a composite case was created for the purpose of performing sensitivity analyses. This composite case, Case A, includes the best estimate probability as well as the contribution from the other identified contributors to severe accidents. Results from the sensitivity analyses can then be compared to Case A to determine the relative impact that variations in input parameters have on the overall estimate of the frequency of the Postulated Sequence.

5.4 SENSITIVITY EVALUATION

There are uncertainties associated with any probabilistic model. The purpose of this section is to address selected uncertainties that may have a substantial impact on the calculated frequency of SFP cooling under the postulated scenario. The sensitivity cases are used to explore those quantitative inputs, modeling, or completeness issues that could vary substantially and influence the results.

The general topics for the sensitivity evaluation include the following:

- Level 1 and 2 Severe Accident Frequencies
- System capabilities during severe accidents
- Plant Configuration
- Operator Actions during severe accidents
- Seismic response capabilities
- Exothermic reactions probability

The sensitivity cases related to each of these are discussed in the following text. It is noted that although the seismic accident sequence sensitivities are discussed last in this section, they are used in the evaluation of each of the other sensitivity cases identified above.

Level 1 and 2 Severe Accident Frequencies (Cases A.1 and A.2)

The frequency of a severe accident (core damage) caused by internal events that can lead to core damage and containment failure or bypass has an uncertainty associated with it. The calculated core damage frequency for SHNPP has an estimated uncertainty characterized by a lognormal distribution with an Error Factor of approximately 6 based on comparison with the NRC analysis in NUREG-1150.

This is characterized as follows:

Characterization	Internal Events Frequency (per yr)
95% Upper Bound	2.5E-5
Mean(1)	7.66E-6
Median	4.22E-6
5% Lower Bound	7.02E-7

Two sensitivity studies are used to demonstrate the impact of considering variations in the quantitative inputs to the SFP analysis by using the 5% and 95% bounds for these inputs. These two sensitivity cases are discussed below.

Varying the accident sequence frequencies for Steps 1 and 2 of the ASLB Order can be performed by changing the frequencies to their 5% (Case A.1) or 95% (Case A.2) bounds. See Tables 5-2 and 5-3 for the lower and upper bound evaluation results, respectively. Note an exception to the above characterization of the uncertainty range is for an ISLOCA. The ISLOCA frequency upper bound has been estimated at approximately 50 times its point estimate value as an upper bound rather than approximately 3 for other sequences.

<u>System Capabilities During Severe Accidents (Case A.3)</u>

The performance of systems during severe accidents can be degraded by the adverse environmental conditions. For the Base Case evaluation, the systems exposed to adverse environments have had their performances adversely impacted in most sequences. In one protected area, equipment is assigned a high probability of reliable operation. The one area is the 6.9KV switchgear rooms to provide offsite power to the demineralized water pumps. If a pessimistic modeling of the 6.9KV switchgear is included in the probabilistic analysis, then an estimate of the impact can be made in Case A.3. (see Table 5-4).

Plant Configuration (Case A.4)

The plant configuration that is not explicitly modeled in the probabilistic model is the possibility that gates either between A and B SFPs or between C and D SFPs are in place.

The Base Case evaluation is performed with the specified SFP configuration. In particular, the probability that the gates are installed in their normal configurations as described in Appendix A is assigned a value of 1.0. However, there is a small probability that maintenance could be required that would result in installation of Gates 3 or 4 for the A and B SFPs or Gates 7 or 9 for the C and D SFPs.

The effects of these configuration changes are to isolate the following:

- SFP A from SFP B - Gate 3 or 4.
- SFP C from SFP D - Gate 7 or 9.

However, the probability of these configurations is estimated to be no larger than 1% of the time for each gate. A sensitivity can be performed to demonstrate the effect of having the gates installed for the maximum of 1% of the time. The sensitivity inputs are:

- Gate 3 or 4 installed 1% of the time.
- Gate 7 or 9 installed 1% of the time.

(1) Mean frequency of core damage and containment failure or bypass calculated in the SHNPP Level 1 and 2 PSA for internal events.

- The time to boil (SFP A) in the worst case is reduced from 20 hours to 6 hours in the worst case.
- The time to uncover fuel (SFP A) in the worst case could be reduced from 6 days to approximately 2 days.
- The HEP for action to align the makeup systems could become higher because of the reduced time available to take effective action. Upon reviewing the HRA, it is found that the HEP increases by a factor of less than 1.25 for each of critical actions (or 1.56 for coupled actions).

The result of these changes can be compared with the Base Model. The Base Model calculation was for the frequency of a radionuclide release from the SFPs with the subject gates always removed; i.e., the frequency of radionuclide release for the 2% of the time that the gates are in place is not increased.

Base Case

- $F^{B}_{Release}$ = 0.98 * X + 0.02 * X = 1.0X

Where X = the calculated frequency of radionuclide release with the Base Case configuration (Gates Out)

Sensitivity Case with Gates In for 1% of Time in A and B and 1% of Time in C and D

- $F^{S}_{Release}$ = 0.98 * X + 0.01 * Z + 0.01 * Y

Where:

Z = the calculated frequency of radionuclide release with the Gate configuration such that A and B are isolated from each other Z = 1.56 * X, based on increased human error probabilities due to decreased time available to respond effectively.

Y = the calculated frequency of radionuclide release with the Gate configuration such that C and D are isolated from each other Y = 1.56 * X, based on increased human error probabilities due to decreased time available to respond effectively.

- $F^{S}_{Release}$ = 0.98 * X + 0.01 * 1.56 X + 0.01 * 1.56 X
- $F^{S}_{Release}$ = 1.01 X

This indicates that explicit treatment of the gates in the model would result in approximately a 1% increase in the calculated frequency of the SFP fuel being uncovered. The increase is so small because of the small probability of the configuration being present and the relatively small impact on the calculated operating crew and TSC response.

Operator Actions During Severe Accidents (Cases B.1, B.2, B.3, B.4, B.5, B.6)

The human action portion of the analysis is crucial to the Best Estimate characterization of SFP cooling following the postulated severe accidents. This is because human intervention is required to prevent evaporation from the SFP's. In order to address this crucial area of the analysis, there are a series of sensitivity cases that are performed to characterize the human interface. These include the following:

- Explicit TSC Guidance - Case B.1
- Access Compromised for ISLOCA, but with explicit TSC Guidance - Case B.2
- Access Compromised for ISLOCA and Upper Bound ISLOCA frequency, but with explicit TSC Guidance - Case B.3
- All human actions included at pessimistic failure probabilities - Case B.4
- Reasonable probability estimates of human actions - Case B.5
- Pessimistic impacts of the on-site radionuclides - Case B.6

Table 5-5 provides the operator action HEP's for cases B.1, B.2, and B.3. These human interface sensitivity cases are described in more detail as follows:

- Case B.1: The use of Best Estimate operator responses given the condition that explicit guidance for the TSC exists to support the alignment of makeup sources at an early time frame. There is some uncertainty regarding the timing and cues that would trigger the use of non-proceduralized and proceduralized actions in aligning makeup to the SFPs. The largest impacts are those associated with the internal events analysis. Overall a reduction of a factor of two in the calculated frequency of uncovering spent fuels is found if more explicit guidance is provided to the TSC than currently exists. [Table 5-6 provides the results.]

- Case B.2: This is the same as Case B.1, except an additional consideration is included that prohibits access to the 216' El North of the FHB due to radiation levels under ISLOCA conditions. The ISLOCA is one of the severe accidents that is being explicitly quantified consistent with the postulated sequence in the Board's Order. The ISLOCA sequence is calculated to be of low frequency and have potentially high offsite consequences. It also has severe effects on the RAB and FHB environments. These severe effects include adverse effects on personnel access and equipment operability which in this sensitivity case preclude the successful mitigation of the event by access to the FHB within 96 hours.

The sensitivity indicates that if the ISLOCA causes a sufficiently high dose to preclude access to the FHB 216'El North, it results in a 30% increase in the internal events contribution to the loss of effective spent fuel makeup. [Table 5-6 provides the results.]

- Case B.3: The same as Case B.2, except that the frequency of the ISLOCA core damage sequences uses the upper bound estimate of ISLOCA frequency which is slightly larger than the older (out of date) IPE analysis. The frequency of ISLOCA has a noteworthy impact on the frequency of the interruption of effective spent fuel cooling. The increase in ISLOCA frequency by a factor of 50 (upper bound) coupled with the limited access to the FHB assumption will lead to a total frequency of loss of SFP cooling and makeup of approximately 4.8E-7/yr. This means that the ISLOCA frequency and its effect on personnel access are some of the key inputs to the quantitative assessment of risk. [Table 5-6 provides the results.]

- Case B.4: All the human actions included in the post containment failure time frame for SFP boiling mitigation are set to 0.1 (or to 1.0 if they are 1.0 in the Base Case). This does not apply to responses where the containment has not failed. Table 5-7 summarizes the HEP's that are used in this sensitivity case. Table 5-8 provides the results of this sensitivity case.

- Case B.5: All the human actions included in the post containment failure time frame for SFP boiling mitigation are set to IE-3 (or to 1.0 if they are 1.0 in the Base Case). Table 5-9 summarizes the HEP's that are used in this sensitivity case. Table 5-10 provides the results of this sensitivity case.

- Case B.6: This sensitivity case represents a pessimistic evaluation of the radionuclide release from the containment. It includes the following:

Accident Type/ Containment Failure Mode	Probability		Site Access for Restoration of Makeup (OPERZOFFST)
	No Access to FHB 286'El.	No Access to FHB 216'El.N.	
SGTR	1.0	0.0	0.5
ISLOCA	1.0	1.0	0.5
Containment Isolation Failure	1.0	0.0	0.5
Early Containment Failure	1.0	1.0	0.5
Late Containment Failure	0.0	0.0	0.5

The purpose of this sensitivity case is to examine under pessimistic meteorological conditions and conservative plume modeling whether effective actions can be taken to provide mitigation. The results indicate that inhibiting access to critical areas of the FHB, the intake structure, and the cooling tower basin due to external plume effects could result in an increase in the frequency of the SFP evaporation and uncovering of the spent fuel by a factor of 4.7. Table 5-11 provides the results of this sensitivity case.

<u>Exothermic Reaction Probabilities (Case C.1)</u>

- <u>Case C.1</u>: A Best Estimate analysis would treat the SFP exothermic reaction in Pools C and D in a way that minimizes the maximum error that can occur given our current state of knowledge for this event. Analytic evidence indicates the possibility of such a reaction under high decay heat and high burnup. Spent fuel in SFP C and D, however, is not consistent with these preconditions. Therefore, the probability of 0.5 would be justified because it will minimize the maximum error that can be made.

Table 5-12 summarizes the results of this evaluation using the Case A characterization of Steps 1-6.

Seismic Response Capabilities

There are also a number of seismic related sensitivities performed to demonstrate the approximate uncertainty bounds on the seismic accident sequences.

Section 4.2 has identified the sensitivity cases to be discussed here. They are summarized in Table 5-13 and are discussed individually regarding their seismic contribution and also how they relate to the other sensitivity cases, A.1 to A.4, B.1 to B.6, and C.1.

The initial statement regarding seismic uncertainties is that the seismic hazard function and the equipment fragilities have substantial uncertainties. This model uses a curve fit to the mean hazard curve (the basis of the best estimate analysis) developed by Lawrence Livermore National Laboratory. Because of the lognormal uncertainty distribution, the mean hazard curve results in the best estimate being close to the upper bound. The lower bound is substantially below the mean. The upper bound hazard curve ranges from a factor of 1.9 times higher than the mean curve for low magnitude seismic events to a factor of 1.7 for high magnitude seismic events. Increasing only the seismic hazard frequency accordingly in each seismic interval results in a seismic induced frequency of spent fuel uncovery of 1.48E-7/yr. Therefore, even with the upper bound hazard curve the sequence frequency does not increase substantially from the best estimate.

On the other hand, the lower bound hazard curve ranges from a factor of 0.15 times lower than the mean curve for low magnitude seismic events to a factor of 0.01 for high magnitude seismic events. Using the lower bound seismic hazard frequency accordingly in each seismic interval results in a spent fuel uncovery frequency of 2.29E-9/yr. Therefore, the use of the lower bound hazard curve produces a substantial reduction in the sequence frequency (more than a factor of 35) compared with the Base Case seismic evaluation.

In addition to the variations in the hazard curve, ten separate seismic sensitivity cases were defined and quantified. The base case seismic assessment and seismic sensitivity case results are summarized in Table 5-13. Each of the ten sensitivity cases are described below.

- (Sensitivity Case S.1) Finer Division of Seismic Hazard Curve: This sensitivity case divides the SHNPP seismic hazard curve into 16 intervals (15 intervals between 0 and 1.5g, and one interval for >1.5g) instead of the Base Case 7 intervals. This sensitivity case tests the impact on the quantitative results from the analysis approach of dividing the seismic hazard curve into discrete intervals, quantifying the risk of each magnitude interval, and then integrating the results. Seismic PSAs typically divide the seismic hazard curve into approximately a half dozen intervals – the approach taken in the Seismic Base Case. Sixteen intervals is a comparatively fine division of the curve. The first fifteen intervals are 0.1g wide (e.g., 0 – 0.1, 0.1 – 0.2, 0.2 – 0.3, etc.) and the final interval is defined as >1.5g.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 7.42E-8/yr (a 15% reduction in frequency compared to the Seismic Base Case). This reduction is not unexpected; the coarser the division of the seismic hazard curve, the more conservative will be the final integrated results.

- (Sensitivity Case S.2) No Extrapolation Beyond NUREG-1488 Hazard Curve: This sensitivity case defines the final seismic magnitude range as >1.0g instead of the Seismic Base Case >1.5g. In the Seismic Base Case, the point at which the FHB is assumed to structurally fail given the seismic shock (and, thus, fall outside the bounds of this analysis) is 1.5g. However, NUREG-1488 only supplies frequency estimates for seismic events up to 1.0g; as such, a case may be made for defining >1.0g as the final magnitude range and assuming that seismic events beyond this are very low likelihood and highly likely to result in FHB failure.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 5.14E8-/yr (a 40% reduction in frequency compared to the Seismic Base Case). This reduction is not unexpected; high magnitude seismic events, although low in frequency, impact the quantitative results due to high component and structural fragilities at such g levels.

- (Sensitivity Case S.3) Less Conservative Uncertainty Distribution for Seismic Fragilities: This sensitivity case employs less conservative randomness and uncertainty parameters (0.30 and 0.30); respectively in the fragility calculations instead of the Base Case values of 0.40 and 0.40. This sensitivity case tests the impact on the quantitative results from the estimated randomness and uncertainty in the component and structural fragility calculations. Randomness and uncertainty parameters used in seismic PSAs are typically in the 0.20 to 0.40 range. In certain cases, values as low as 0.10 – 0.20 (e.g., offsite power transformers) and as high as 0.50 – 0.70 (e.g., relay chatter failures) are used. The Seismic Base Case employs 0.40 and 0.40 as a suitably conservative set of values. This sensitivity case uses 0.30 and 0.30 to represent a less conservative set of values.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 5.40E-8/yr (a 37% reduction in seismic induced accident sequence frequency compared to the Seismic Base Case). This reduction is not unexpected; all other issues being equal, the tighter the assumed uncertainty around the estimated seismic capacities, the lower are the calculated fragilities.

- (Sensitivity Case S.4) Seismic Capacities Increased Approximately 25%: This sensitivity case employs higher component and structural seismic capacities than used in the Seismic Base Case. The Seismic Base Case uses component and structural capacities estimated based on review of similar components in other seismic PSAs and knowledge of the SHNPP plant. This sensitivity case tests the impact on the quantitative results given the possibility that the selected capacities used in the assessment are conservative. A factor of approximately 1.25 was assumed in this sensitivity to indicate the comparative level of conservatism existing in the selected capacities of the Seismic Base Case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 3.65E-8/yr (a 58% reduction in frequency compared to the Seismic Base Case). This reduction is not unexpected; all other issues being equal, the higher the estimated seismic capacities, the lower are the calculated fragilities.

- (Sensitivity Case S.5) Seismic Capacities Decreased Approximately 25%: This sensitivity case employs lower component and structural seismic capacities than used in the Seismic Base Case. The Seismic Base Case uses component and structural capacities estimated based on review of similar components in other seismic PSAs and knowledge of the SHNPP plant. This sensitivity case tests the impact on the quantitative results given the possibility that the selected capacities used in the assessment are non-conservative. A factor of approximately 0.75 was assumed in this sensitivity to indicate a comparative level of non-conservatism that may be postulated to exist in the selected capacities of the Seismic Base Case.

As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 1.62E-7/yr (1.9 times the Seismic Base Case). This increase is not unexpected; all other issues being equal, the lower the estimated seismic capacities, the higher are the calculated fragilities.

- (Sensitivity Case S.6) More Conservative Early Containment Failure Probability: This sensitivity case employs a higher early containment failure probability than used in the Seismic Base Case. The Seismic Base Case uses a conditional (upon core damage) early containment failure probability of 3.76E-2 based on review of the current SHNPP PSA results. The 3.76E-2 value is the most conservative value of the assessed core damage scenarios. This sensitivity case tests the impact on the quantitative results from a higher early containment failure probability. An approximate factor of 3 is applied to the Seismic Base Case value, resulting in a nominal early containment failure probability of 0.10 for use in this sensitivity case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 1.12E-7/yr (a 30% increase in frequency compared to the Seismic Base Case). This increase is not unexpected because early containment failure directly impacts the human error probabilities associated with providing cooling to the SFPs.

- (Sensitivity Case S.7) More Conservative Human Error Probabilities: This sensitivity case employs higher human error probabilities than used in the Seismic Base Case. The Seismic Base Case generally employs conservative human error probabilities (e.g., 1.0AC power recovery failure probability, 1.0 manual containment isolation failure probability). This sensitivity case applies a conservative element across the board to all human errors. Human error probabilities less than 0.1 are set to 0.1, and human error probabilities greater than or equal to 0.1 are left at the Seismic Base Case value.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 1.46E-7/yr (1.7 times the Seismic Base Case).

This increase is not unexpected; human error probabilities play a key role in the assessed spent fuel failure frequency.

- (Sensitivity Case S.8) Less Conservative Human Error Probabilities: This sensitivity case employs less conservative human error probabilities for selected human interfaces in the Seismic Base Case. The Seismic Base Case generally employs conservative human error probabilities (e.g., 1.0 AC power recovery failure probability, 1.0 manual containment isolation failure probability). This sensitivity case reduces the 1.0 failure probabilities to 0.5 for the following selected actions:

  - AC Power Recovery Failure
  - Containment Manual Isolation Failure
  - Fire Hose Alignment Failure Given Early Containment Failure
  - Fire Hose Alignment Failure Given Containment Isolation Failure

  All other human error probabilities are left at the Seismic Base Case value.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 3.86E-8/yr (a 55% decrease in frequency compared to the Seismic Base Case). This decrease is not unexpected; human error probabilities play a key role in the assessed spent fuel failure frequency.

- (Sensitivity Case S.9) Overall Pessimistic Case: This sensitivity case employs all the attributes of Sensitivity Cases 5, 6, and 7. This sensitivity case is aptly described as the overall pessimistic case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 3.43E-7/yr (4 times the Seismic Base Case).

- (Sensitivity Case S.10) Overall Optimistic Case: This sensitivity case employs all the attributes of Sensitivity Cases 1, 2, 3, 4 and 8. This sensitivity case is aptly described as the overall optimistic case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 2.06E-9/yr (a 97% decrease in frequency compared to the Seismic Base Case).

5.5 SENSITIVITY RESULTS

Table 5-14 summarizes the results of the sensitivity cases performed to characterize the degree of uncertainty in the quantitative evaluation of the Postulated Sequence. As discussed in Section 5.3, the best estimate of the probability of the Postulated Sequence is best represented by the probability calculated for internal events alone. This is due to the level of uncertainty associated with the state of the technology for the calculation of external event and shutdown contributions. The sensitivity of the analysis to various input parameters, is shown relative to a composite Base Case, Case A. The sensitivity cases then used a composite frequency as well, and are compared to Case A to demonstrate the sensitivity of the probability estimate to the various input parameters. The results, therefore, include the contributions to the Postulated Sequence from internal, seismic, fire and shutdown events. The results make use of the appropriate seismic sensitivity cases.

Figure 5-1 provides a histogram comparison of the sensitivity results using the composite totals from internal, seismic, fire, and shutdown events. This figure also compares the results with the NRC surrogate safety goal for severe accidents leading to core damage (i.e., 1E-4/reactor year). In addition, the frequency cited in Appendix B of this report as "remote and speculative" is also shown for reference (i.e., 1E-6/year).

Figure 5-1 includes estimated upper and lower bounds on the evaluation based on the comparison of the sensitivity cases. These bounds should be interpreted to represent an approximation to the 90% confidence interval within which the frequency may lie.

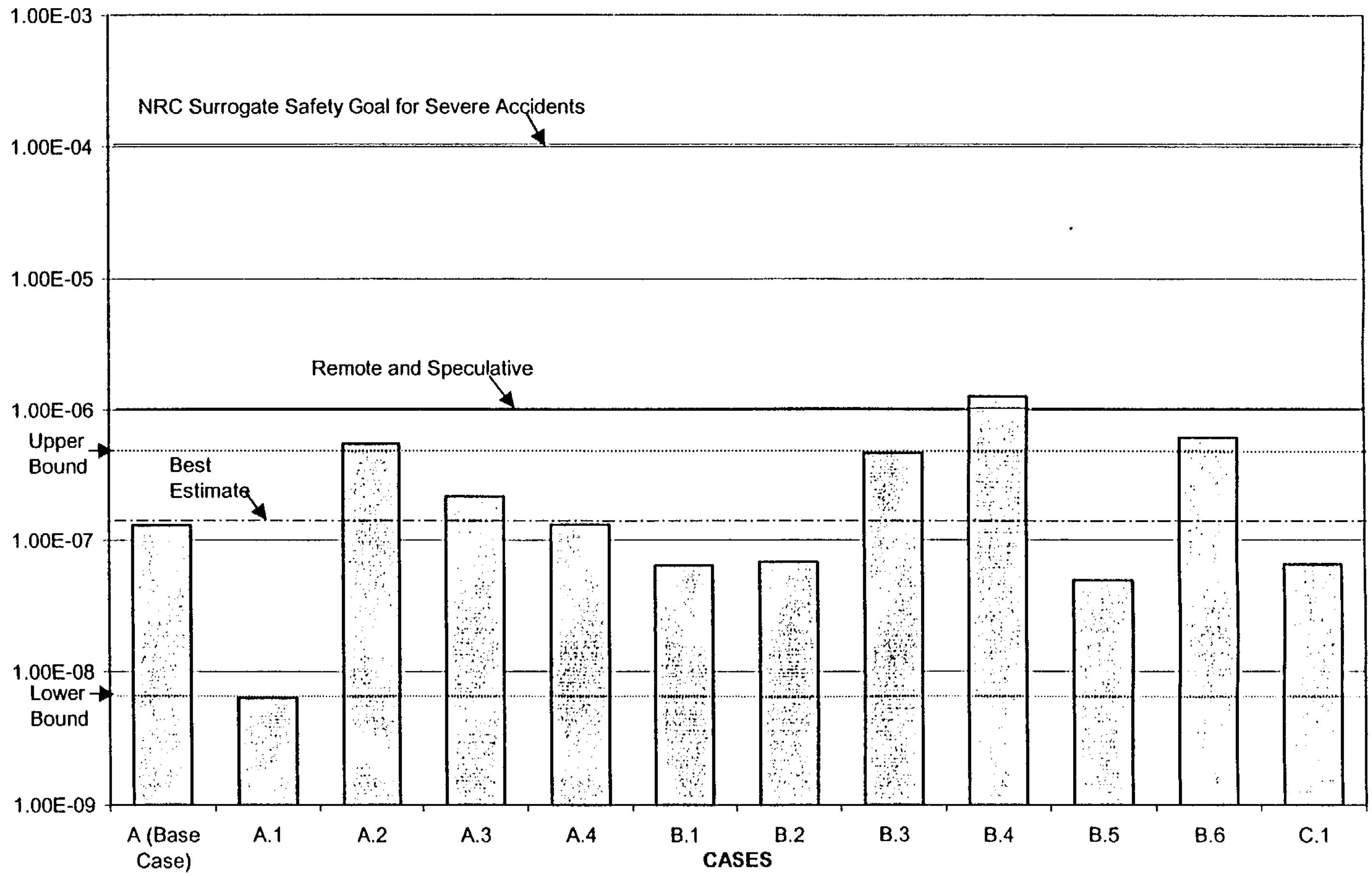

Figure 5-1 Summary of Sensitivity Cases to Demonstrate the Range of Uncertainty

Table 5-1

SHNPP SFPAET RESULTS
BEST ESTIMATE ACCIDENT SEQUENCE FREQUENCIES

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	7.44E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.44E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.31E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	9.77E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.59E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	1.15E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.43E-08
Total Internal Events Contribution		7.67E-06	2.65E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	7.98E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	2.86E-09
Total Fire Events Contribution		9.80E-07	2.94E-09
Total Seismic Contribution		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.45E-08

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-2

SHNPP SFPAET RESULT LOWER BOUND
ACCIDENT SEQUENCE FREQUENCIES (CASE A.1)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	0.0	0.0
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.4E-07	3.16E-10
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.4E-07	3.07E-10
LG-ISOL	LARGE ISOLATION FAILURE	7.0E-09	9.01E-11
SM-ISOL	SMALL ISOLATION FAILURE	1.7E-08	2.34E-10
EARLY	EARLY CONTAINMENT FAILURE	2.9E-09	2.89E-10
LATE	LATE CONTAINMENT FAILURE	3.9E-07	1.30E-09
Total Internal Events Contribution		7.0E-07	2.54E-09
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-10	7.98E-12
LATE	LATE CONTAINMENT FAILURE	9.77E-08	2.86E-10
Total Fire Events Contribution		9.80E-08	2.94E-10
Total Seismic Contribution (Case S.10)			2.1E-09
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	5.0E-08	1.45E-09

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-3

SHNPP SFPAET RESULTS UPPER BOUND ACCIDENT SEQUENCE FREQUENCIES (CASE A.2)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	5.0E-7	3.73E-08
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	5.1E-06	1.12E-08
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	4.9E-06	1.07E-08
LG-ISOL	LARGE ISOLATION FAILURE	2.5 E-07	3.22E-09
SM-ISOL	SMALL ISOLATION FAILURE	6.1E-07	8.40E-09
EARLY	EARLY CONTAINMENT FAILURE	1.0E-07	3.66E-09
LATE	LATE CONTAINMENT FAILURE	1.4E-05	4.68E-08
Total Internal Events Contribution		2.55E-05	1.21E-07
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-.08	7.98E-10
LATE	LATE CONTAINMENT FAILURE	9.77E-06	2.86E-08
Total Fire Events Contribution		9.80E-06	2.94E-08
Total Seismic Contribution (Case S.9)		--	3.4E-7
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	2.0E-06	5.80E-08

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-4

SHNPP SFPAET RESULTS FOR PESSIMISTIC MODELING OF 6.9KV SWITCHGEAR SURVIVABILITY[1] (CASE A.3)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[2]	Output from SFPAET[3]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	4.8E-09
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.05E-08
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.01E-08
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	3.08E-09
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	8.06E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	2.67E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	3.47E-08
Total Internal Events Contribution		7.67E-06	7.4E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	2.19E-10
LATE	LATE CONTAINMENT FAILURE	9.77E-07	6.75E-09
Total Fire Events Contribution		9.80E-07	6.97E-09
Total Seismic Contribution (Base Case)[4]		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	5.38E-08

(1) Set the Demineralized Water Pumps to 1.0

(2) CDF with containment failure, bypass, or containment isolation failure (per year).

(3) Frequency of the loss of effective water cooling to the spent fuel (per year).

(4) Seismic event involves Loss of Offsite Power; therefore no effect of the Normal 6.9KV Power Switchgear.

Table 5-5

SHNPP SFPAET SENSITIVITY RESULTS

Basic Event	Description	Base Case	Case B.1, B.2, B.3
OPERDALNPB	Operators Fail To Align DW To The Unit 1 or Unit 2 FPCCS Cleanup Subsystem	1.90E-02	9.5E-3
OPER-TSC-E	TSC Fails to Take Pre-emptive Action for Early Failures	4.6E-03	2.4E-3
OPERPALNN1	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	6.2E-2	1.1E-3
OPERPALNN2	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	1.00E+00	2.5E-1
OPER-TSC-L	TSC fails to take PRE-emptive Action for Late Failures	2.4E-3	1.4E-3

Table 5-6

SHNPP SFPAET SENSITIVITY RESULTS: CASE B.1, B.2, B.3

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Base Output[1] from SFPAET	Sensitivity Case B1[1]	Sensitivity Case B2[1]	Sensitivity Case B3[1]
Internal Events					
ISLOCA	INTERFACING SYSTEMS LOCA	7.44E-10	7.44E-10	9.0E-09	4.03E-07
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	3.44E-09	1.57E-09	1.57E-09	1.57E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	3.31E-09	1.51E-09	1.51E-09	1.51E-09
LG-ISOL	LARGE ISOLATION FAILURE	9.77E-10	7.99E-10	7.99E-10	7.99E-10
SM-ISOL	SMALL ISOLATION FAILURE	2.59E-09	2.16E-09	2.16E-09	2.16E-09
EARLY	EARLY CONTAINMENT FAILURE	1.15E-09	1.15E-09	1.15E-09	1.15E-09
LATE	LATE CONTAINMENT FAILURE	1.43E-08	8.12E-09	8.12E-09	8.12E-09
Total Internal Events Contribution		2.65E-08	1.60E-08	2.43E-08	4.18E-07
Fire Induced Events					
EARLY	EARLY CONTAINMENT FAILURE	7.98E-11	8.35E-11	8.35E-11	8.35E-11
LATE	LATE CONTAINMENT FAILURE	2.86E-09	1.30E-09	1.30E-09	1.30E-09
Total Fire Events Contribution		2.94E-09	1.38E-09	1.38E-09	1.38E-09
Total Seismic Contribution (Case S.8)		8.65E-08	3.88E-08	3.88E-08	3.88E-08
Shutdown Events					
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	1.45E-08	7.62E-09	7.62E-09	7.62E-09

(1) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-7
SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: PESSIMISTIC HEP'S

New Basic Event	Case B.4	Description	OP-116 Step
OPERDALNPB	0.1	Operators Fail To Align DW To The Unit 1 FPCCS Cleanup Subsystem	8.4
OPERDALNPB	0.1	Operators Fail To Align DW To The Unit 2 FPCCS Cleanup Subsystem	8.4
OPER-1CLBA	0.1	Operators Fail To Cross Tie Unit 1 FPCCS Pump Train B To Heat Exchanger A	N/A
OPER-2CLBA	0.1	Operators Fail To Cross Tie Unit 2 FPCCS Pump Train B To Heat Exchanger A	N/A
OPERPALNN1	0.1	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	N/A
OPER-GATE1	1	Operators Fail To Deflate Gate 1 Seals	N/A
OPER-GATE2	1	Operators Fail To Deflate Gate 2 Seals	N/A
OPER-GATE3	1	Operators Fail To Deflate Gate 3Seals	N/A
OPER-GATE4	1	Operators Fail To Deflate Gate 4 Seals	N/A
OPER-GATE5	1	Operators Fail To Deflate Gate 5 Seals	N/A
OPER-GATE6	1	Operators Fail To Deflate Gate 6 Seals	N/A
OPER-GATE7	1	Operators Fail To Deflate Gate 7 Seals	N/A
OPER-GATE9	1	Operators Fail To Deflate Gate 9 Seals	N/A
OPER-GATES	1	Operators Fail To Remove Bulkhead Gates	8.27
OPERPALNN2	1.0	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	N/A
OPERPALNN3	1	Operators Fail To Use Water From The NSW System In The WPB To Makeup To The SFP	N/A
OPER-OFFST	0.1	Operators Fail To Use Portable / Off-Site Resources For Makeup To The SFPs	N/A
OPER-PROCD	0.1	Procedures To Maintain SFP Inventory Are Inadequate	All

Table 5-7
SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: PESSIMISTIC HEP'S

New Basic Event	Case B.4	Description	OP-116 Step
OPERRALNPC	1	Operators Fail To Align The FPCCS Purification Subsystem To The RWST	8.5
OPER-LOLVL	0.1	Operators Fail To Diagnose Low SFP Levels And / Or Perform Recovery	All
OPER-ESW	0.1	Operators Fail To Open ESW Manual Valves	8.13
OPER-TSC-E	0.1	TSC Fails to Take Pre-emptive Action for Early Failures	NA
OPER-TSC-L	0.1	TSC Fails to Take Pre-emptive Action for Late Failures	NA
OPER-SKIMR	1	Operators Fail To Open The Crosstie Between Units 1 and 4 and 2 and 3 FPCCS Skimmers	NA
OPER-DWXTM	1	Operators Fail To Open DM Crosstie Valve 1SF-203	NA
OPER-START	0.1	OPERATORS FAIL TO MANUALLY START FPCS MOTOR-DRIVEN PUMP	NA
OPERZOFFST	0.1	Operator Fails to Align Offsite Resources to Previously Established Paths	NA
CI-CASE 1	1.1 E-2	Operator Fails to Restore Primary Containment Given Mid Level Operation (Shutdown only)	Tech specs
CI-CASE 2	1.6 E-2	Operator Fails to Restore Primary Containment Given Normal Level Operation (Shutdown only)	Tech specs
OPERATOR ACTIONS GIVEN NO CREDIT IN ANALYSIS			
OPEREALNPA	1	Operator Fails to Align and Initiate ESW to FPCC for Makeup	8.13
OPERMALNPD	1	Operator Fails to Align and Initiate RMWST to FPCC for Makeup	8.26
OPERDALNPE	1	Operator Fails to Align and Initiate Demin Water to FPCC Skimmer for Makeup	8.6
OPERRALNPF	1	Operator Fails to Align and Initiate RWST to FPCCS Cooling Pump for Makeup	8.5
OPERDALNPG	1	Operator Fails to Align and Initiate Demin Water to FPCC Cleanup for Makeup	8.5
OPER-IN-FA	1	Operator Fails to Initiate FPCC Cooling to Pools A and B	N/A
OPER-IN-FC	1	Operator Fails to Initiate FPCC Cooling to Pools C and D	N/A

Table 5-8

SHNPP SFPAET RESULTS (CASE B.4) PESSIMISTIC HEPs

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from level 1 and 2 Quantification[(1)]	Output from SFPAET[(2)]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	3.99E-09
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.73E-07
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.66E-07
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	8.46E-09
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.22E-08
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	8.17E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	4.98E-07
Total Internal Events Contribution		7.67E-06	9.98E-07
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	6.87E-10
LATE	LATE CONTAINMENT FAILURE	9.77E-07	1.66E-07
Total Fire Events Contribution		9.80E-07	1.17E-07
Total Seismic Contribution (Case S.7)			1.46E-07
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.44E-07

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-9

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: REASONABLE HEP's

New Basic Event	BASE case	Description	OP-116 Step
OPERDALNPB	IE-03	Operators Fail To Align DW To The Unit 1 FPCCS Cleanup Subsystem	8.4
OPERDALNPB	IE-03	Operators Fail To Align DW To The Unit 2 FPCCS Cleanup Subsystem	8.4
OPER-1CLBA	IE-03	Operators Fail To Cross Tie Unit 1 FPCCS Pump Train B To Heat Exchanger A	N/A
OPER-2CLBA	IE-03	Operators Fail To Cross Tie Unit 2 FPCCS Pump Train B To Heat Exchanger A	N/A
OPERPALNN1	IE-03	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	N/A
OPER-GATE1	1	Operators Fail To Deflate Gate 1 Seals	N/A
OPER-GATE2	1	Operators Fail To Deflate Gate 2 Seals	N/A
OPER-GATE3	1	Operators Fail To Deflate Gate 3Seals	N/A
OPER-GATE4	1	Operators Fail To Deflate Gate 4 Seals	N/A
OPER-GATE5	1	Operators Fail To Deflate Gate 5 Seals	N/A
OPER-GATE6	1	Operators Fail To Deflate Gate 6 Seals	N/A
OPER-GATE7	1	Operators Fail To Deflate Gate 7 Seals	N/A
OPER-GATE9	1	Operators Fail To Deflate Gate 9 Seals	N/A
OPER-GATES	1	Operators Fail To Remove Bulkhead Gates	8.27
OPERPALNN2	1	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	N/A
OPERPALNN3	1	Operators Fail To Use Water From The NSW System In The WPB To Makeup To The SFP	N/A
OPER-OFFST	1.00E-03	Operators Fail To Use Portable / Off-Site Resources For Makeup To The SFPs	N/A
OPER-PROCD	1.00E-03	Procedures To Maintain SFP Inventory Are Inadequate	All
OPERRALNPC	1	Operators Fail To Align The FPCCS Purification Subsystem To The RWST	8.5

Table 5-9

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: REASONABLE HEP's

New Basic Event	BASE case	Description	OP-116 Step
OPER-LOLVL	1.00E-03	Operators Fail To Diagnose Low SFP Levels And / Or Perform Recovery	All
OPER-ESW	1.00E-03	Operators Fail To Open ESW Manual Valves	8.13
OPER-TSC-E	1.00E-03	TSC Fails to Take Pre-emptive Action for Early Failures	NA
OPER-TSC-L	1.00E-03	TSC Fails to Take Pre-emptive Action for Late Failures	NA
OPER-SKIMR	1	Operators Fail To Open The Crosstie Between Units 1 and 4 and 2 and 3 FPCCS Skimmers	NA
OPER-DWXTM	1	Operators Fail To Open DM Crosstie Valve 1SF-203	NA
OPER-START	2.00E-05	OPERATORS FAIL TO MANUALLY START FPCS MOTOR-DRIVEN PUMP	NA
OPERZOFFST	1.00E-03	Operator Fails to Align Offsite Resources to Previously Established Paths	NA
OPERATOR ACTIONS CURRENTLY MODELED AS GUARANTEED FAILURE			
CI-CASE 1	1.1 E-2	Operator Fails to Restore Primary Containment Given Mid Level Operation (Shutdown only)	Tech specs
CI-CASE 2	1.6 E-2	Operator Fails to Restore Primary Containment Given Normal Level Operation (Shutdown only)	Tech specs
OPERMALNPD	1	Operator Fails to Align and Initiate RMWST to FPCC for Makeup	8.26
OPERDALNPE	1	Operator Fails to Align and Initiate Demin Water to FPCC Skimmer for Makeup	8.6
OPERRALNPF	1	Operator Fails to Align and Initiate RWST to FPCCS Cooling Pump for Makeup	8.5
OPERDALNPG	1	Operator Fails to Align and Initiate Demin Water to FPCC Cleanup for Makeup	8.5
OPER-IN-FA	1	Operator Fails to Initiate FPCC Cooling to Pools A and B	N/A
OPER-IN-FC	1	Operator Fails to Initiate FPCC Cooling to Pools C and D	N/A

Table 5-10

SHNPP SFPAET RESULTS (CASE B.5): REASONABLE HEPs

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input CDF from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	3.99E-11
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.57E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.51E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	8.45E-11
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.22E-10
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	7.13E-11
LATE	LATE CONTAINMENT FAILURE	4.28E-06	4.27E-09
Total Internal Events Contribution		7.67E-06	7.77E-09
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	5.63E-12
LATE	LATE CONTAINMENT FAILURE	9.77E-07	9.88E-10
Total Fire Events Contribution		9.80E-07	9.94E-10
Total Seismic Contribution (Case S.8)			3.90E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.44E-09

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-11

SHNPP SFPAET RESULT FOR HIGH ON-SITE RADIATION DUE TO CONSERVATIVE CHI/Q ACCIDENT SEQUENCE FREQUENCIES (CASE B.6)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[(1)]	Output from SFPAET[(2)]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	9.97E-09
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.36E-08
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.24E-08
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	6.51E-09
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	1.81E-08
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	3.14E-08
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.03E-07
Total Internal Events Contribution		7.67E-06	2.51E-07
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-10	2.95E-09
LATE	LATE CONTAINMENT FAILURE	9.77E-08	1.69E-08
Total Fire Events Contribution		9.80E-08	1.99E-08
Total Seismic Contribution (Case S.9)			3.40E-07
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.60E-08

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-12

SHNPP SFPAET RESULTS FOR ASSESSMENT OF SENSITIVITY TO EXOTHERMIC REACTION PROBABILITY ACCIDENT SEQUENCE FREQUENCIES (CASE C.1)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[(1)]	Output from SFPAET[(2)]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	3.70E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.70E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.70E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	4.90E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	1.30E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	5.80E-10
LATE	LATE CONTAINMENT FAILURE	4.28E-06	7.20E-09
Total Internal Events Contribution		7.67E-06	1.37E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	4.00E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	1.40E-09
Total Fire Events Contribution		9.80E-07	1.50E-09
Total Seismic Contribution (Special Case)		-	4.30E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	7.30E-09

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-13
SUMMARY OF SEISMIC ASSESSMENT QUANTITATIVE SENSITIVITY CASES

		Seismic Hazard Curve		Seismic Fragility Parameters								Human Interfaces										
														Fire Hose Align HEP			Demin Align HEP					
Sensitivity Case	Case Description (1)	# Seis. Mag. Intervals	Magnitude of Final Seismic Range	BETA(r), BETA(u)	EDG Am	Ess. SWGR Am	PCIV Am	DFP Am	FHB Flooding Am	Offsite Infrastructure Am	Early Containment Failure Probability	AC Recovery Failure Prob.	PCIV Manual Isolation	Early Cont. Failure	Cont. Isol. Failure	Late Cont. Failure	Early Cont. Failure	Cont. Isol. Failure	Late Cont. Failure	Fire Truck Hook-Up HEP	Portable Pump/Gen Hook-Up HEP	Spent Fuel Uncovery Frequency (1/yr)
0	BASE Case	7	>1 5g	0 4,0 4	1 25	1 31	2 00	1 25	1.25	1 00	3 76E-2	1 00	1 00	1 00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	8 65E-08
1	Finer Division of Seismic Hazard Curve	18	>1 5g	0 4,0 4	1 25	1 31	2 00	1 25	1.25	1 00	3 76E-2	1 00	1 00	1 00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	7 42E-08
2	No Extrapolation Beyond NUREG-1488 Hazard Curve	7	>1.0g	0 4,0 4	1 25	1 31	2 00	1 25	1.25	1 00	3 76E-2	1 00	1 00	1 00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	5 14E-08
3	Less Conservative Uncertainty Distribution for Seismic Fragilities	7	>1 5g	0.3,0 3	1 25	1 31	2 00	1 25	1 25	1 00	3 76E-2	1 00	1 00	1 00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	5 40E-08
4	Seismic Capacities Increased Approximately 25%	7	>1 5g	0.4,0 4	1.50	1.65	2.50	1.50	1.50	1.25	3 76E-2	1 00	1 00	1 00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	3 65E-08
5	Seismic Capacities Decreased Approximately 25%	7	>1 5g	0 4,0 4	1.00	1.00	1.50	1.00	1.00	0.75	3 76E-2	1 00	1 00	1 00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	1 62E-07
6	More Conservative Early Containment Failure Probability	7	>1 5g	0 4,0 4	1.25	1.31	2 00	1.25	1.25	1.00	1.00E-1	1 00	1.00	1.00	1 00	0 062	0 10	0 019	0 019	1 00	0 05	1 12E-07
7	More Conservative Human Error Probabilities	7	>1.5g	0 4,0 4	1.25	1.31	2.00	1.25	1.25	1 00	3 76E-2	1.00	1 00	1 00	1 00	0.10	0 10	0.10	0.10	1 00	0.10	1.46E-07
8	Less Conservative Human Error Probabilities	7	>1 5g	0 4,0 4	1.25	1 31	2 00	1 25	1.25	1.00	3.76E-2	0.50	0.50	0.50	0.50	0 062	0 10	0 019	0 019	1 00	0 05	3 86E 08
9	Overall Pessimistic Case	7	>1 5g	0 4,0 4	1.00	1.00	1.50	1.00	1.00	0.75	1.00E-1	1.00	1 00	1 00	1 00	0.10	0 10	0.10	0.10	1 00	0.10	3 43E-07
10	Overall Optimistic Case	11 (Note 2)	>1.0g	0.3,0.3	1.50	1.65	2.50	1.50	1.50	1.25	3 76E-2	0.50	0.50	0.50	0.50	0 062	0 10	0 019	0 019	1 00	0 05	2 06E-09

NOTES:
(1) Shaded cells indicate parameter changes with respect to the BASE Case.
(2) Ten seismic hazard intervals between 0 0 and 1 0g, and one interval for >1 0g.

Table 5-14

SENSITIVITY CASE RESULTS

Sensitivity		Factor of Change Compared with Case A	Comments on Results
Case No.	Description		
A	Case A	1.3E-7/yr	This includes the best estimate contributions to the probability of the Postulated Sequence from the internal, seismic, fire, and shutdown analyses.
A.1	Lower Bound for Accident Frequencies (Steps 1 and 2) (Uses Case S.10 for seismic)	20 Reduction	Lower Bound estimate on the input accident frequency state in turn results in a substantial decrease in the SFP undesirable end state frequency estimates.
A.2	Upper Bound for Accident Frequencies (Steps 1 and 2) (Uses Case S.9 for seismic)	4.27 increase	Use of Upper Bound estimates on the inputs lead to a factor of 4 increase in the frequency SFP undesirable end state frequency.
A.3	Pessimistic Assessment of 6.9KV Switchgear Survivability (Uses Base Case for seismic)	1.67 increase	The impact of switchgear survivability for use of offsite power affects the internal events, shutdown and fire contributions. The use of a pessimistic assumption leads to a modest increase in the frequency of the undesirable end state.
A.4	Upper Bound Estimate for Installation of Gates Between A and B or Between C and D	1.01 increase	Essentially no impact on the Base Case evaluation.

Table 5-14

SENSITIVITY CASE RESULTS

Sensitivity		Factor of Change Compared with Case A	Comments on Results
Case No.	Description		
B.1	Written TSC Guidance Provided (Uses Case S.8 for seismic)	2.0 reduction	Written guidance regarding actions to be taken under severe accident conditions is calculated to lead to a reduction of approximately a factor of 2 in the frequency of SFP undesirable end state.
B.2	Access During ISLOCA Precluded (Uses Case S.8 for seismic)	1.8 reduction	Access to the FHB under ISLOCA conditions are found to have minimal impact on the assessed frequency when the Best Estimate ISLOCA frequency is used. Results are dominated by the TSC Guidance addition.
B.3	B.2 Plus Higher ISLOCA Frequency (Uses Case S.8 for seismic)	3.58 increase	When the upper bound ISLOCA frequency AND no access to the FHB are included in the quantitative model, it is found that the frequency of the undesirable end state for the SFP is found to increase by a factor of 3.6.
B.4	Degraded Human Response for all POST Containment Failure Actions (Uses Case S.7 for seismic)	9.85 increase	Because of the strong interface with operating crew actions, the calculated end state frequency is sensitive to changes in the HEPs

Table 5-14

SENSITIVITY CASE RESULTS

Sensitivity		Factor of Change Compared with Case A	Comments on Results
Case No.	Description		
B.5	Human Errors Are Set to 1E-3 to characterize a reasonable response to severe accidents (Except Guaranteed Failure Cases) (Uses Case S.8 for seismic)	1.61 reduction	Further reductions in the post containment failure HEPs from those used in the Base model have a relatively small impact on the results.
B.6	Accessibility Based on Worst Case Site Deposition with Chi/Q model (Uses Case S.9 for seismic)	4.6 increase	Radionuclide releases that are postulated to contaminate the site under worst case assumptions could lead to a substantial increase in the frequency of the undesirable SFP condition.
C.1	Estimate of Exothermic Reaction in SFP if water has evaporated	2 reduction	The exothermic reaction conditional probability is essentially a straight multiplier on the results. Therefore, a conditional probability that minimizes the maximum error, 0.5, results in a reduction in the undesirable end state of a factor of 2.

Section 6
CONCLUSIONS

6.1 OVERVIEW

A comprehensive PSA has been performed in response to the Postulated Sequence of events contained in the ASLB's August 7, 2000 Memorandum and Order. The PSA establishes the best estimate, given the current state of knowledge and technology, of the overall probability of the chain of seven events (Postulated Sequence) at SHNPP following the commencement of SFP C and D operation. The chain of seven events in the Postulated Sequence are as follows:

1. A degraded core accident
2. Containment failure or bypass
3. Loss of all spent fuel cooling and makeup systems
4. Extreme radiation doses precluding personnel access
5. Inability to restart any pool cooling or makeup systems due to extreme radiation doses
6. Loss of most or all pool water through evaporation
7. Initiation of an exothermic oxidation reaction in pools C and D.

This analysis has directly responded to the ASLB Order and establishes the probability for the specific scenario outlined by this Postulated Sequence. Furthermore, because the Postulated Sequence is focused on the ability of plant personnel to respond to the outlined events, this analysis did not consider off-site consequences associated with the scenario.

The seven steps of the Postulated Sequence are described in the following text; some related steps are discussed together.

Steps 1 and 2

A degraded core accident occurs and containment fails or is bypassed. Core damage sequences for which the containment is failed or bypassed as a result of internal, seismic, fire, and shutdown events are addressed in the quantitative assessment. The best estimate evaluation is judged to be best characterized by the internal events contribution. (See Section 4)

Step 3

Loss of all spent fuel pool cooling and makeup systems were considered as a result of the accident sequence and probabilistically, due to random or human-induced failures. (See Section 4, Appendices A, C and E).

Steps 4 and 5

For all sequences identified in Steps 1 and 2, radiation levels were calculated for specific areas in which access would be necessary in order to respond to Step 3. Consideration of the adverse impacts of extreme radiation on both personnel access and equipment survivability were then included in the probabilistic assessment. In addition, adverse environments due to high temperature or high humidity were deterministically assessed and included in the probabilistic model. (See Section 4, Appendices, A, C and E).

Step 6

Loss of most or all pool water through evaporation were then considered. To assess the probability of this step, a comprehensive analysis of the SFPs was conducted. The analysis considered the specific characteristics of the SFPs at SHNPP, as well as the potential methods available for injection of water in the event of the Postulated Sequence. A probabilistic assessment of the potential for the loss of SFP water through evaporation due to the loss of cooling and makeup systems was included. (See Section 4)

Step 7

Initiation of an exothermic oxidation reaction in pools C and D was then evaluated to determine whether it could be estimated probabilistically. Determining a best estimate probability for this step in the Postulated Sequence was difficult, given the state of knowledge related to this phenomenon. With the limited time and resources available to respond to the Postulated Sequence, this analysis assumes that the initiation of a self sustaining exothermic oxidation reaction in SFPs C and D occurred with a probability of 1.0, if the previous six steps had led to the evaporation of water from the SFPs. CP&L has addressed qualitatively how unlikely such an exothermic oxidation reaction would be in SFPs C and D. (See Affidavit of Robert K. Kunita.) Therefore, the assigned conditional failure probability of 1.0 is conservative.

The effort to respond to the ASLB Order involved the formation of an analysis team (13 Team Members) and a direct link to key CP&L staff. The CP&L staff provided detailed calculations (including the Level 1 and 2 SHNPP PSA), system descriptions, interviews with operating personnel, and procedure interpretations. The team effort included:

- multiple SHNPP site visits to confirm the as-built design and crew response;
- an independent peer review of the inputs to the evaluation, including the Level 1 and 2 PSA; and,
- an independent review of this analysis.

The methods chosen to evaluate each of the seven steps and arrive at a best estimate of the overall probability are characteristic of methods that have been used to perform past nuclear power plant PSAs. Where possible, this analysis relied on the results from the SHNPP Level 1 and Level 2 PSA. The specific method employed for each type of potential severe accident contributor that was evaluated varied according to the type of event being considered and the current state of technology:

Potential Severe Accident Contributor		Methodology Utilized
Internal Events	-	Full PSA methodology
Fire	-	Full PSA methodology for dominant sequences
Seismic	-	Approximate method
Shutdown	-	Generic assessment based on similar PWRs
Other	-	Determined to have negligible contribution

The SHNPP PSA (Level 1 and 2 Internal Events) was subjected to an independent peer review process as part of this evaluation. The review determined that the SHNPP PSA was robust, comprehensive, and consistent with the state-of-the-technology for such probabilistic assessments in the industry. The SHNPP PSA for internal events is fully supportive of risk-informed applications, even in cases where the absolute frequency of the accident sequences is required to support the application. The peer review also confirmed the finding of the SHNPP PSA (Level 1 and 2 Internal Events) that the plant meets the NRC Safety Goals and their subsidiary objectives (i.e., Core Damage Frequency and Large Early Release Frequency). In addition, the peer review confirmed that there are no unusual contributors to core damage frequency or containment failure.

6.2 CONCLUSIONS

Determination of the type of severe accidents that could result in the chain of events in the Postulated Sequence was the first step in this analysis. The analysis concluded that degraded core conditions with containment failure or bypass could result from a number of different postulated accident scenarios, which can be discussed under the following general categories of events differentiated by mode of operation:

A. At-Power

- Internal Events
- Internal Flood
- Seismic Induced
- Fire Induced
- Other

B. Shutdown

- Shutdown

This conclusion led to the separation of these severe accidents into two main subgroups, 1) Internal Events and 2) External Events and Shutdown. As discussed earlier in this report, the state of knowledge regarding the quantitative assessment of risk at nuclear power plants is best developed for assessing the risk due to internal events. It was therefore concluded that the best estimate of probability of the Postulated Sequence would be best determined by consideration of internal events. Following the determination of the best estimate probability for internal events, external events and shutdown events were evaluated to determine whether these events alter the conclusion reached based on the internal events assessment. These sensitivity analyses demonstrated that the best estimate probability that was determined was reasonable.

The results of the best estimate assessment for sequences initiated by internal events indicated that the loss of effective SFP cooling has an annual occurrence probability of 2.65E-8. Compared with other rare and accepted risks in life, this can be considered remote and speculative. The annual occurrence probability of the Postulated Sequence is, for example, considerably less than the probability of the recurrence of the ice age or the probability of a meteor strike creating worldwide havoc. (See Appendix B).

The conclusion from the external events and shutdown analysis is that the uncertainties associated with these sequences are sufficiently large that several conservatisms have been incorporated in the modeling. These conservatisms potentially result in inflated point estimate calculations. Therefore, while the point estimate contribution due to seismic initiated events is higher than for internal events, it is judged not to alter the conclusions reached based on the internal events analysis, i.e., that the postulated sequences of events can be considered "remote and speculative."

Table 6-1 is a summary table of the analysis results for the best estimate of the annualized probability of evaporation of SFP water and the uncovering of spent fuel from internal events, fire induced events, seismic events and shutdown events. The frequency for each event type is listed in the "output" column of Table 6-1. The internal event contribution directly responds to the questions regarding the Postulated Sequence presented in the ASLB Order, except it treats the time during the evaporation of water below the top of the fuel as inconsequential to the analysis and treats the probability of an exothermic reaction as equal to 1.0.

Fire induced events and shutdown events have a probability even lower than that estimated for internal events, and thus support the conclusion that the probability of the Postulated Sequence is below regulatory significance. The seismic contribution was calculated to be somewhat higher than the probability calculated for internal events. However, the Postulated Sequence requires that such a seismic event would have to be large enough to cause core damage and containment failure or bypass, and yet not damage the SFPs so as to preclude Step 6. Thus, the seismic evaluation is considered a "conservative" estimate not a "Best Estimate" as specified in the ASLB Question.

There are three main conclusions that can be drawn from the PSA applied to the chain of seven steps , and they can be qualitatively summarized based on the quantitative results and sensitivity evaluations:

1. The postulated chain of events is beyond the plant design basis.
2. The frequency of the Postulated Sequence is considered extremely low and is "remote and speculative".
3. The addition of SFPs C and D to SHNPP does not increase the frequency of the scenario. In fact, the plant modifications associated with the commissioning of SFPs C and D actually decrease the frequency of uncovering spent fuel at SHNPP. This is related to the new plant configuration which adds a viable makeup pathway under nearly all postulated accidents.

6.3 CONSERVATISMS

Despite all prudent attempts to create a best estimate evaluation, there remain some potential residual conservatisms in the quantification. Among these conservatisms are the following:

- Containment basemat failure has been treated in a manner that always causes a release into the RAB. The exact basemat failure locations are not defined in the Level 2 PSA. Therefore, this assumption has been made because of the lack of adequate information.

- A substantial fraction of the containment does not interface with the RAB. However, the dominant failure modes for containment appear to be at locations where RAB impacts cannot be ruled out. Therefore, all containment failures are assumed to impact the RAB environment.

- The SFP boil off time is taken to be the minimum it can be, given the plant configuration and the times at which freshly discharged spent fuel could be introduced into the A and B SFPs.

- The seismic evaluation is subject to large uncertainty and is believed to be a conservative bound because of the assumptions of :

  -- Loss of site power with no opportunity for recovery

  -- Complete dependence of failures of similar components

  -- The early containment failure probability used in the seismic evaluation is the worst case found for any plant damage state. This is likely too conservative when applied to the seismic initiated sequences involving station blackout.

- Many motor operated pumps are located in the RAB or the FHB and are exposed to various degrees of harsh conditions, depending on their spatial relationship to the location of the primary containment failure. These pumps may fail to operate if an adequate room environment is not maintained.

An increase in the ambient temperature, due to loss of room cooling or due to primary containment failure, is the main concern. A conservative approach is taken by assuming that components fail if the room temperature exceeds the manufacturer recommended value. However, in the case of pump motors, the failure is more a function of time at temperature rather than simply exceeding a temperature limit. Therefore, continued pump operation may be likely even for temperatures exceeding manufacturer specified warranty values.

The pump motors may also fail due to moisture intrusion. The humid environment in the pump areas following primary containment failure would likely result in moisture intrusion in the CCW and ESW Booster Pump motors that could potentially result in shorted or grounded circuits. The CCW and ESW Booster Pumps are not credited with continuous operability following containment failure scenarios.

- The treatment of containment isolation failures into the RAB in the base model assumes that access to the RAB and FHB operating deck (286' Elevation) is not available. This is conservative relative to the deterministic calculations performed to support accessibility. The deterministic calculations indicate that the FHB is not affected by the Containment Isolation failure. Therefore, there is a slight conservatism in the current model. This is a conservatism, but it does not substantially reduce the calculated frequency. It also does not change the conclusions of the study.

- Air cooling of spent fuel that has low decay heat levels may be an effective cooling method (based on existing NRC National Laboratory calculations). However, this mode of cooling was not quantitatively credited in this Base Case PSA and the probability of a self-sustaining exothermic oxidation reaction in the event of uncovering a substantial portion of the spent fuel (Step 7) was assumed to be 1.0. A best estimate probability would require a detailed heat balance evaluation of the SFP, which is beyond the scope of this evaluation. The qualitative analysis of the temperatures that might be reached in SFPs C and D recognizing the heat rates of the fuel that would be stored (particularly if limited to 1.0 MBTU per hour) that was performed by CP&L would suggest that the conditional probability of Step 7 would be considerably less than 1.0.

Table 6-1

SHNPP SFPAET RESULTS BASE CASE
ACCIDENT SEQUENCE FREQUENCIES (CASE A)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1&2 Quantification[(1)]	Output from SFPAET[(2)]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	7.44E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.44E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.31E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	9.77E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.59E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	1.15E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.43E-08
Total Internal Events Contribution		7.67E-06	2.65E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	7.98E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	2.86E-09
Total Fire Events Contribution		9.80E-07	2.94E-09
Total Seismic Contribution		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.45E-08

(1) CDF with containment failure, bypass, or containment isolation failure(per yr).

(2) Frequency of the loss of effective water cooling to the spent fuel(per yr).

Section 7

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Appendix A

SPENT FUEL POOLS AND ASSOCIATED EQUIPMENT

Appendix A

SPENT FUEL POOLS AND ASSOCIATED EQUIPMENT

This Appendix provides a description of the key features of the Shearon Harris fuel handling building (FHB) and spent fuel pools (SFPs) and the systems that perform important functions associated with the SFPs. The appendix includes the following:

- Description of the location of the SFPs in the FHB
- Description of the SFPs
- Description of the SFP cooling and support systems
- Description of makeup methods for adding water to the SFP
- Description of the instrumentation used to monitor the SFP and cue any operator actions to the maintenance of adequate fuel cooling

A.1 FUEL HANDLING BUILDING

The Harris Fuel Handling Building is atypical of many nuclear power plants because of its large size. The FHB was constructed to accommodate a four unit site. Therefore, the size and compartmentalization of the building makes its response to a loss of cooling potentially different than many other sites. This feature of the Harris FHB has been explicitly represented in the deterministic calculations of post containment failure accident sequences.

Fuel Handling Building

The Shearon Harris Nuclear Power Plant (SHNPP) FHB is situated to the east of the Unit 1 power block and to the north of the Waste Processing Building (WPB). Its south wall abuts the WPB. Its east wall abuts the Unit 1 Reactor Auxiliary Building (RAB). Its west wall abuts structures that were to have been the Unit 4 and Unit 3 RABs. Its north wall does not abut any structures.

Figures A-1 through A-4 show the various elevations of the FHB.

The FHB consists of four levels plus the roof:

- 337 ft elevation – Roof. Notable components located on the roof include the RAB / FHB HVAC exhaust stack. Access to the FHB roof is from the adjacent RAB roof.

- 286 ft elevation – Main operating floor and top of all SFPs and transfer canals. Notable components located on this elevation of the FHB include: the fuel handling bridge; Fuel Pool Cooling and Cleanup System (FPCCS) skimmer subsystem skimmers (23) floating on the surface of the SFPs and canals; demineralized water system manual valve stations (19) along the west and east walls; FPCCS skimmer subsystem manual valves located along the tops of the SFPs and canals in service valve boxes; seven fire hose stations, each containing a 1.5" fire hose; FHB control panels FP-9 and FP-10 along the east wall; and the FHB 10 ton auxiliary crane. In addition, two 480 VAC General Service Buses (1-4A102 and 1-4B102) are located in a separate room on the south end of this elevation; this room may only be entered from the outside, from doors located off of the WPB roof. The FHB operating floor may be accessed through doors D893 and D894 in the southwest wall from the WPB stairwell. "Tornado" door D892 leads into this same stairwell airlock area from the FHB roof. There are two stairwells and a freight elevator in the north end of the FHB. The elevator and one of the stairwells go to the railroad bay at elevation 261. The second stairwell provides access to rooms in the northern ends of the 261 ft elevation, the 236 ft elevation and the 216 ft elevation.

- 261 ft elevation (site grade level) – Fuel unloading area (rail access bay) on the north end and a ventilation equipment room (with an attached demineralizer room on its south end) on the south end. Notable components located in the ventilation equipment room (room FH6) on this elevation of the FHB include: normal FHB HVAC and emergency exhaust equipment; 480 VAC motor control centers MCC-1&4A33-SA and MCC-1&4B33-SB (in mechanical equipment sub-room FH7); 480 VAC motor control centers 1-4A1021, 1-4A1022, 1-4B1021 and 1-4B1022; and the FPCCS purification subsystem demineralizers. Access to the ventilation equipment room is through combination double doors / single door D119 in the east wall from the RAB 261 ft elevation. Access to the demineralizer room is either through an open passageway from the south end of the ventilation equipment room or directly through a single door in the east wall from the RAB 261 ft elevation. Access to the railroad bay is from the outside through a large, airtight sliding door on the north end; from a stairwell and an elevator from the 286 ft elevation of the FHB; from the outside through air-tight double man doors to the right of the railroad door; or, from the outside through "tornado" door D3312 in the east wall.

- 236 ft elevation – This elevation of the FHB is comprised of three distinct areas: A room at the south end of the building that does not contain any equipment considered in the SFP cooling or makeup analysis; an equipment area in the central portion of the building; and, a room at the north end of the building. Key components located on this elevation of the FHB in the central equipment room include: FPCCS skimmer subsystem pumps, filter, strainers and demineralizers; FHB control panels FP-7 and FP-8 and associated instrument racks; and FPCCS cooling subsystem pumps, heat exchangers (cooled by component cooling water) and strainers. Access to this room is through either double doors D6500 or adjacent single door D650 from the 236 ft elevation of the RAB in the east wall, or through a single "tornado" door in the west wall from the fabrication shop at the 236 ft elevation (an area that was to have been the Unit 3 RAB). The North 236 ft elevation contains access to that elevation from exterior to the FHB and also access to the North 216 ft elevation.

- 216 ft elevation – Two completely separated compartments (North and South) containing: four (4) FPCCS purification subsystem pumps; demineralized water cross-tie valves 1SF-201 (South 216 ft.) and 2SF-201 (North 216 ft.); FHB floor drains and equipment drains sumps and sump pumps (North and South); FHB HVAC condensate recirculation transfer pump and tank (South room only); FPCCS filter backwash pumps and tanks (North and South); and component cooling water system transfer pump and holdup tank (North room only).

  Access to the South room is through single door D725 in the East wall near the South end or a double door in the east wall near the north end from the 216 ft elevation of the RAB.

Access to the North room is from: (a) the FHB northeast stairway via the 286 ft elevation of the FHB; (b) down the same stairway after entering the North end of the FHB at the 236 ft elevation through "tornado" door D3312 from the safety meeting room in what was to have been the Unit 3 RAB; or, (c) from the 236 ft elevation North end area via a ladder stored at that location without requiring access to the stairwell.

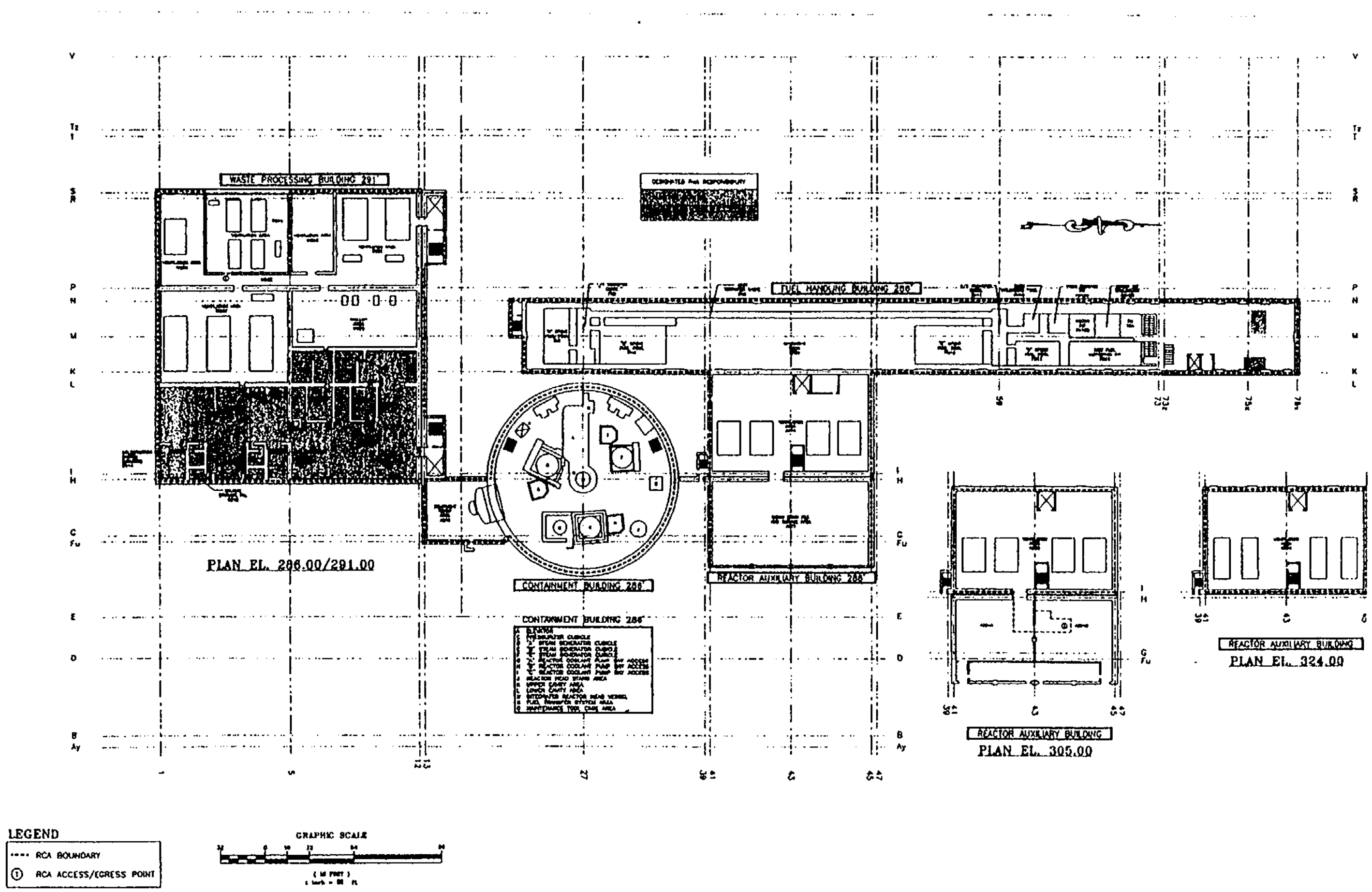


Figure A-1 Elevation 286' of RAB and Operating Deck Level of FHB

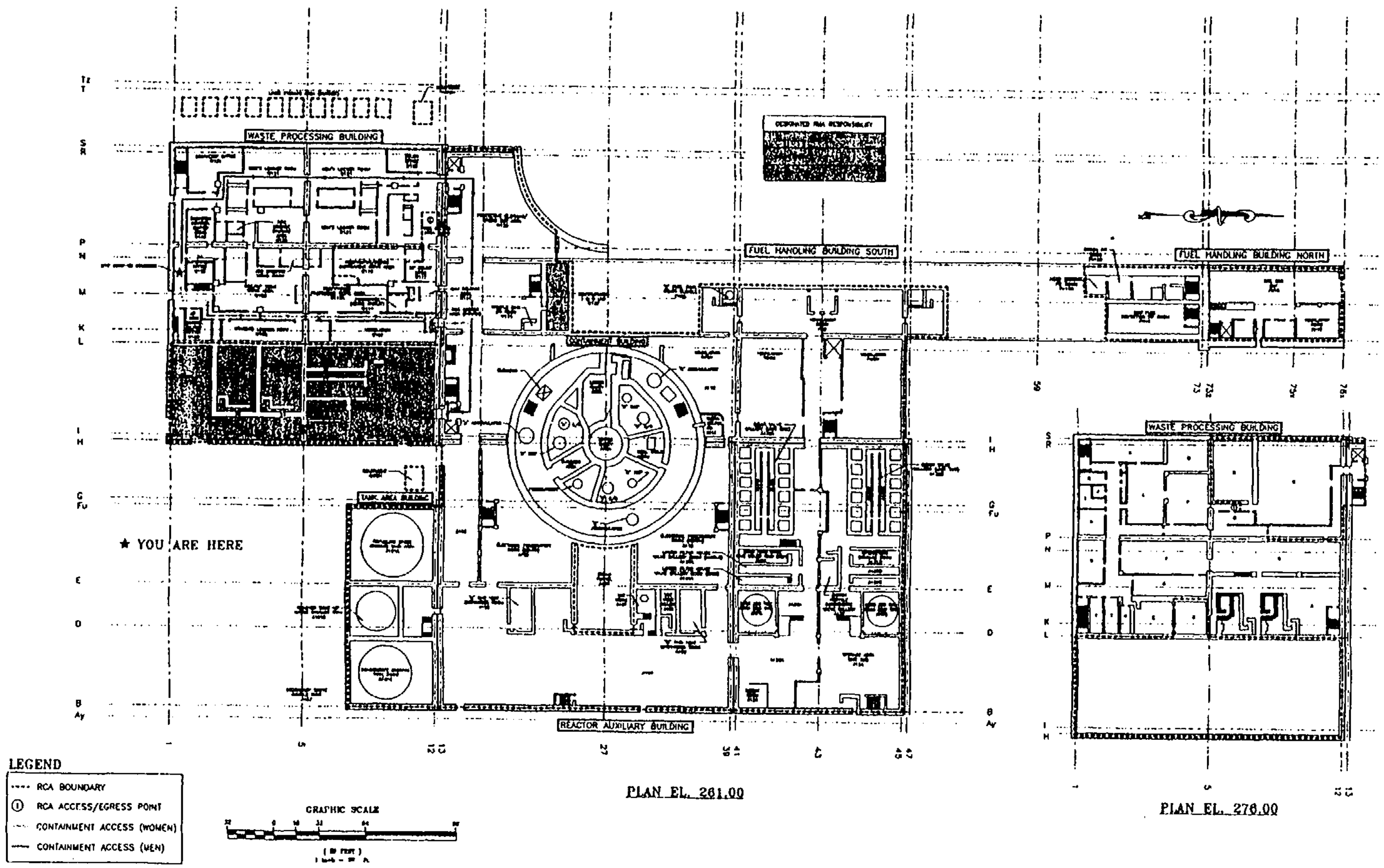


Figure A-2 Elevation 261' of FHB and RAB

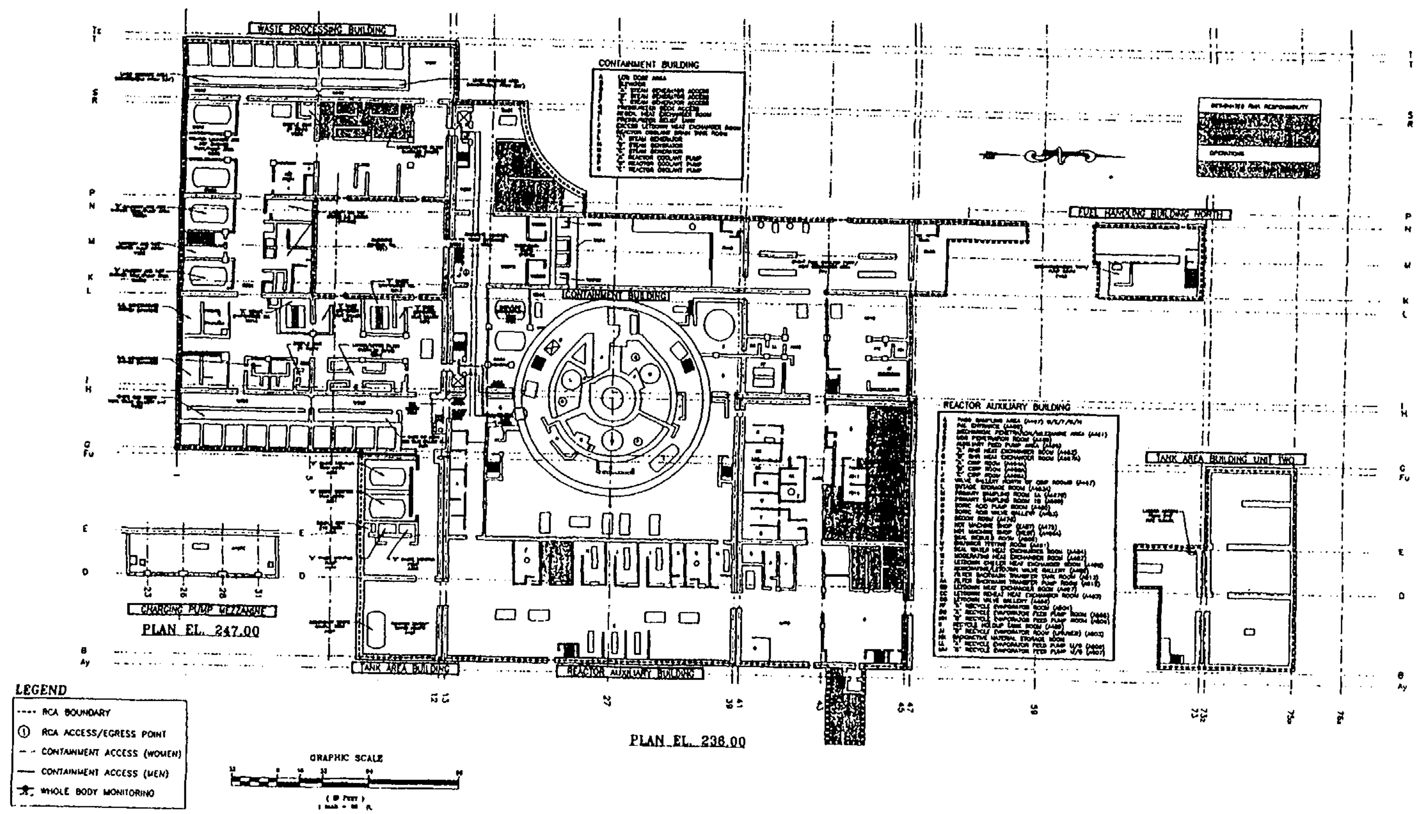
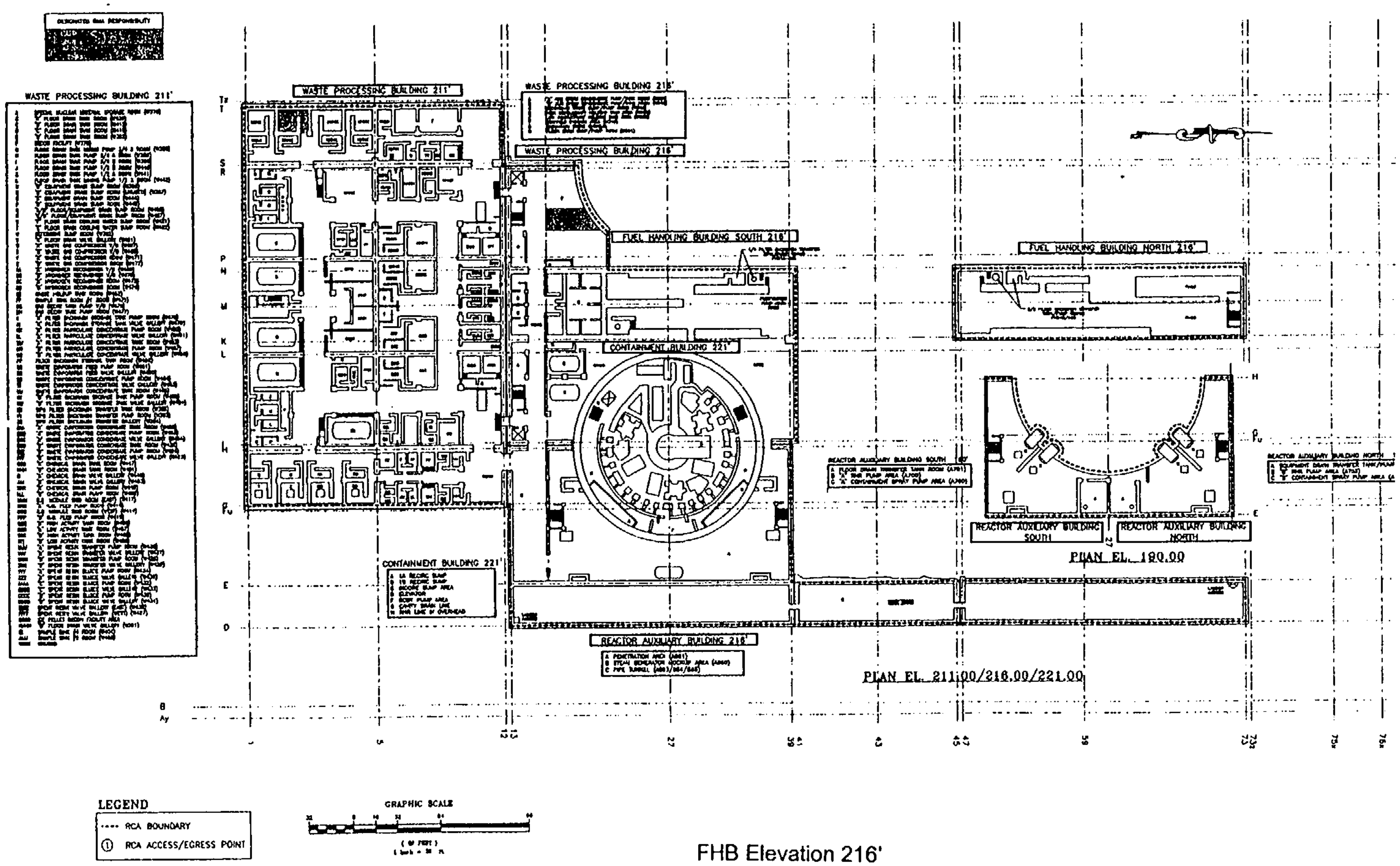


Figure A-3 Elevation 236' of RAB and FHB



FHB Elevation 216'

Figure A-4 FHB Elevation 216'

A.2 SPENT FUEL POOLS

A.2.1 Fuel Pools

The FHB contains five main pools. The south end of the FHB contains the new fuel pool (Pool "A") and a spent fuel pool (Pool "B"). The north end of the FHB contains two spent fuel pools (Pools "C" and "D") and the spent fuel shipping cask loading pool (Cask Loading Pool). These five pools are tied together by 3 interconnected canals: the Main Transfer Canal, the South Transfer Canal and the North Transfer Canal.

The four SFPs and the Cask Loading Pool are reinforced concrete structures with stainless steel liners. The bottoms of the four SFPs are at elevation 246.00 ft. Normal water level in the SFPs is maintained at 284.5 ft. The bottom of the Cask Loading Pool is at elevation 240.00 ft. Normal water level in this pool is maintained at 284.5 ft, consistent with the SFPs.

Draining or siphoning of the pools via piping or hose connections to the pools or the canals is precluded by the location of the penetrations, limitations on hose length, and the termination of piping penetrations flush with the liner. Main Control Room and local alarms are provided to alert operators to abnormal pool levels or high temperatures.

A.2.2 Main Transfer Canal

The Main Transfer Canal runs south to north (parallel to the west wall of the FHB) between the northwest corner of the South Transfer Canal and the southwest corner of the North Transfer Canal.

The Main Transfer Canal is a concrete structure with a stainless steel liner. The bottom of the Main Transfer Canal is at elevation 260.00 ft. Normal water level in the canal is maintained at 284.5 ft, consistent with the fuel pools.

A.2.3 South Transfer Canal

The South Transfer Canal runs west to east between Pools A and B. The Fuel Transfer Tube to the SHNPP Unit 1 Containment enters the east end of the South Transfer Canal. The South Transfer Canal is also connected by channels to Pools "A" and "B."

The South Transfer Canal is a concrete structure with a stainless steel liner. The bottom of the South Transfer Canal is at elevation 251.00 ft. Normal water level in the canal is maintained at 284.5 ft, consistent with the fuel pools.

A.2.4 North Transfer Canal

The North Transfer Canal runs west to east between Pool C and Pool D and the Cask Loading Pool. The North Transfer Canal is connected by channels to Pools "C" and "D" and the Cask Transfer Pool.

The North Transfer Canal is a concrete structure with a stainless steel liner. The bottom of the North Transfer Canal is at elevation 251.00 ft. Normal water level in the canal is maintained at 284.5 ft, consistent with the fuel pools.

A.2.5 Isolation Gates

Nine movable bulkhead gates may be used to isolate the five pools from each other:

- Gate 1 (1SF-E001) – Isolates the South Transfer Canal from the Main Transfer Canal.

- Gate 2 (1SF-E002) – Isolates the Main Transfer Canal from Pool "B."
- Gate 3 (1SF-E003) – Isolates the South Transfer Canal from Pool "B."
- Gate 4 (1SF-E004) – Isolates the South Transfer Canal from Pool "A."
- Gate 5 (1SF-E005) – Isolates the North Transfer Canal from the Main Transfer Canal.
- Gate 6 (1SF-E006) – Isolates the Main Transfer Canal form Pool "C."
- Gate 7 (1SF-E007) – Isolates the North Transfer Canal from Pool "C."
- Gate 8 (1SF-E008) – Isolates the North Transfer Canal from the Cask Loading Pool.
- Gate 9 (1SF-E009) – Isolates the North Transfer Canal from Pool "D."

The bulkhead gates are constructed of stainless steel plate and structural steel members. The sides and the bottoms fit into slots in the SFP's canal walls and floors. Inflatable rubber seals are installed in the sides of the bulkhead gates. The seals are inflated by Instrument Air (IA) once the gates are set in place. IA enters each installed gate's seals via a separate line attached with a quick disconnect plug at the top of the gate. Figure A.2-1 is a simplified schematic of the gate locations in the Spent Fuel Pools.

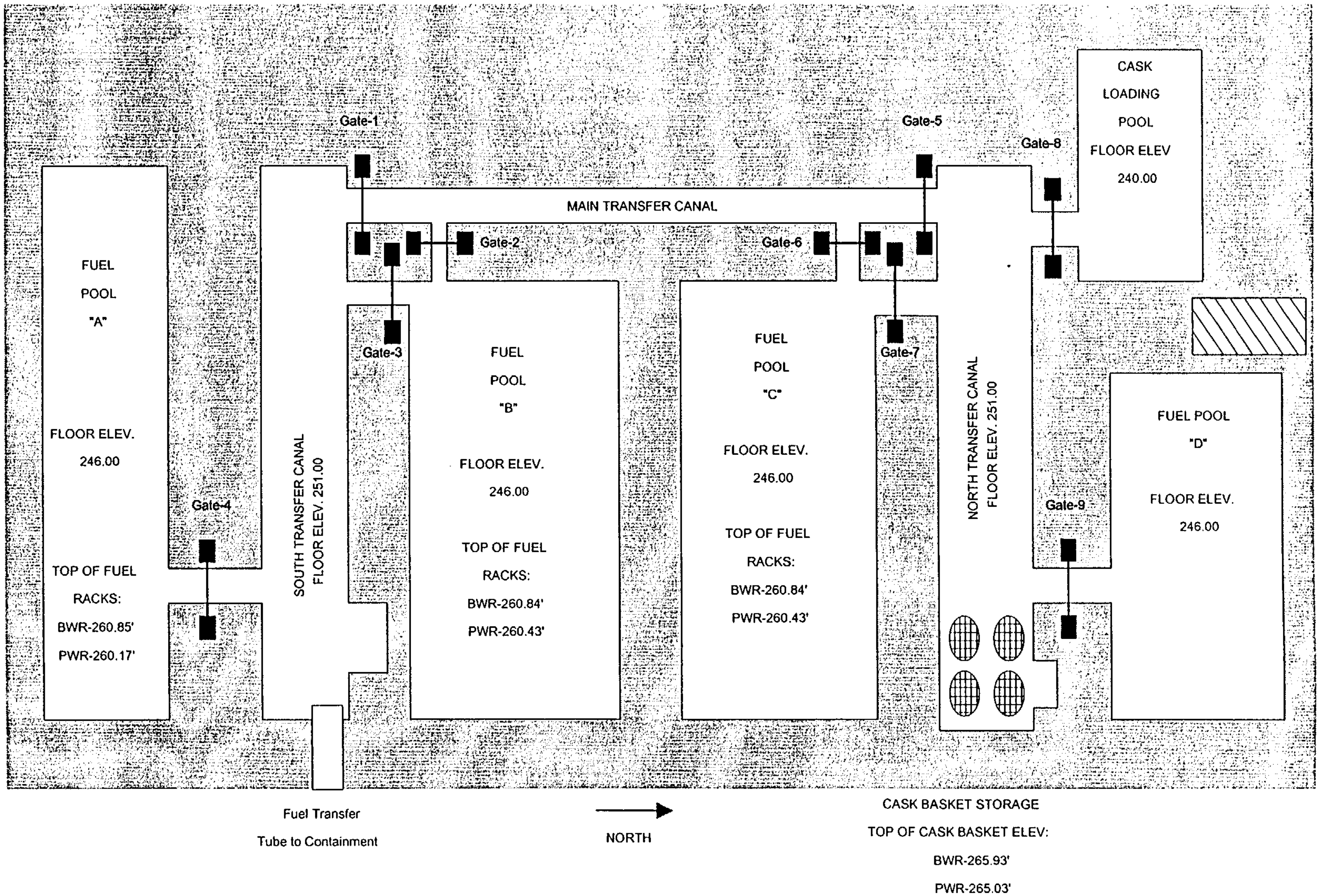


Figure A.2-1 Simplified Schematic of Gate Locations

The gates are moved using the 12-ton FHB Auxiliary Crane (see SHNPP Operations Procedure OP-116 Section 8.27 and Attachment 7). The FHB Auxiliary Crane is powered from 480 VAC MCC 1-4B1022 (fed from General Service Bus 1-4B). The FHB Auxiliary Crane is not available in the event of a loss of off-site power. When they are not in use, the bulkhead gates are placed in dedicated storage areas in the Main Transfer Canal.

Gates (2 and 6) between the pools and the Main Transfer Canal are normally installed. Gates (3, 4, 7, and 9) between the SFPs and the North and South Transfer Canals are not normally installed. Installation and/or removal of gates during an emergency is estimated to require approximately 60 to 90 minutes per gate. Removal of gates in the event of a loss of SFP cooling is not procedurally required. In the case where makeup water from adjacent pools and transfer canals is needed to mitigate a loss of water inventory in a pool, removal of the gates is not required. The pneumatic seal on the gates can be deflated (within a period of minutes) via removal of a quick disconnect fitting or sufficient water can be injected to overflow the gates. Deflating a gate allows water to flow past the gate until an equilibrium water level condition is established. Under these conditions, the exchange and re-equilibration of water between the isolated pool (i.e., gate installed but deflated) and adjacent pools or canals is rapid, and typically occurs on a timescale of minutes. The model built for these analyses contains flag events that may be individually set for each gate; setting a gate's flag event to TRUE would represent that gate being installed.

The gates between SFPs A and B and those between SFPs C and D will be removed under most foreseeable circumstances. There is a very remote potential that maintenance could be required on the pools or transfer canal. This could necessitate installation of the gates for a very short time. This is estimated to occur 1.0% of the time. [Eric McCartney, 9/29/00]. The percentage of time, on an annual basis, that the spent fuel pools would be operated with the gates removed is summarized as follows:

Estimated Percentage of Time, on an Annual Basis, the Bulkhead Gates Would be Normally Removed from the SHNPP Spent Fuel Pools Subsequent to Operational Use of C and D Pools	
Gate Number	Best Estimate Time Gates Removed [A-1]
1	99% [A-28]
2	1%
3	99%
4	99%
5	1%
6	1%
7	99%
8	99%
9*	99%

* The "normally open" configuration for gate 9 (gate removed 99% of the time) would apply subsequent to placing this pool in service, scheduled for early the next decade. Otherwise, this gate would remain normally closed.

The top of the pools and transfer canals (286 ft) is 10.5 inches above the top of the installed gates [A-2]; i.e., the tops of the installed gates are at an elevation of 285 feet 1 ½ inches (285.125 feet). The normal water level of the SFPs and the canals is 284.5 feet, which is 0.625 feet below the top of the installed gates.

A.2.6 Spent Fuel Pool Configurations

The SFP configuration is such that even with the SFP gates in place there would be communication among pools if makeup flow continues to flood a single pool. The water would overflow the gates, but not overflow out of the pools. This overflow would eventually flood all pools.

The boil off rate for the highest heat rate (SFPs A + B @ 25E+6 Btu/hr pool) is estimated at 52 gpm. Therefore, as long as the makeup exceeds this value all pools can be flooded.

The volume to flood the A + B South Canal + Main Transfer Canal pools from the low level point (284') to the overflow of the pools above the gates is 23,000 gal.

A.3 FUEL POOL COOLING AND HEATUP

A.3.1 Fuel Pool Cooling

The Fuel Pool Cooling and Cleanup System (FPCCS) has two primary purposes. It is designed to maintain water quality by removing particulate and dissolved fission and corrosion products resulting from the spent fuel stored in the pools; it is also designed to remove residual heat generated by the spent fuel stored in the pools and to maintain an adequate water inventory in the pools.

The FPCCS consists of the following three subsystems:

1. FPCCS Cooling Subsystem – Pools "A" and "B" are currently served by a two-loop FPCCS cooling subsystem. Major components in each of these loops include a pump, a heat exchanger and a strainer. The heat exchanger is cooled by the Component Cooling Water (CCW) system in the Reactor Auxiliary Building (RAB). Each of the 4560 gpm horizontal centrifugal pumps are able to be powered from the emergency diesel generators (EDGs) following a

loss of off-site power. Each loop of this cooling system is 100% capacity and is independent of the other loop. The pumps are locally controlled from panels FP-7 and FP-9 located in the FHB.

Pools "C" and "D" will be served by a two-loop FPCCS cooling subsystem identical to the system in pools "A" and "B". Installation of this subsystem is scheduled for completion by the end of 2000; it will, therefore, be fully operational prior to commissioning pools "C" and "D" for spent fuel storage. The proposed modification is adopted in this analysis as present when pools "C" and "D" are operational.

2. FPCCS Cleanup / Purification Subsystem – Pools "A" and "B" are currently served by a two-loop FPCCS cleanup subsystem. Major components in each of these loops include a fuel pool demineralizer, a fuel pool demineralizer filter, a fuel pool and refueling water purification filter and a 325 gpm pump. Each of these pumps is capable of taking suction from the canals, the pools, the Unit 1 refueling cavity in Containment and the RWST via the containment spray (CS) system. The system is operated only as needed.

   Pools "C" and "D" will be served by a two-loop FPCCS cleanup subsystem identical to the system in pools "A" and "B". Installation of this subsystem is scheduled for completion by the end of 2000; it will, therefore, be fully operational prior to commissioning pools "C" and "D" for spent fuel storage.

3. Fuel Pools Skimmer System - Pools "A" and "B" are currently served by a skimmer system that consists of a 385 gpm pump, a strainer and a filter. The system removes any floating debris from the surface of the pools and canals via 15 floating skimmers deployed as follows:

   - Pool "A" 3
   - Pool "B" 5
   - South Transfer Canal 2
   - Main Transfer Canal 2
   - North Transfer Canal 2
   - Cask Loading Pool 1

Pools "C" and "D" will be served by their own FPCCS skimmer subsystem identical to the system in pools "A" and "B". Five skimmers will serve pool "C"; three skimmers will serve pool "D". Installation of this subsystem is scheduled for completion by the end of 2000; it will, therefore, be fully operational prior to commissioning pools "C" and "D" for spent fuel storage. This analysis assumes that the modifications are in service when modeling the pools "C" and "D" FPCCS skimmer subsystem.

A.3.2 Fuel Pool Heatup

Calculations were performed by CP&L to determine the time required to reach boiling temperature and then the additional time required to boil the water to the top of the spent fuel racks for spent fuel pools A and B and for spent fuel pools C and D, with loss of spent fuel pool cooling and no operator action. The results of these calculations are summarized below.

The results of these calculations are summarized below:

Pools	Time to reach boiling temperature	Additional time for water level to reach top of racks	Total time	Makeup required to offset boiling
A and B (Beginning of cycle)	20.57 hours	7.21 days	8.07 days	53.70 gpm
A and B (End of cycle)	38.67 hours	13.56 days	15.17 days	28.57 gpm
C and D (1 MBTU/hr heat load)	384.66 hours	99.99 days	116.02 days	2.15 gpm
C and D (15.6 MBTU/hr heat load)	34.42 hours	8.80 days	10.23 days	33.64 gpm

These calculations did not take credit for any additional cooling or makeup that would be available to the pools.

The cases for which calculations have been performed include the following:

A & B (Beginning of cycle):	This represents a case which involves a fuel core off load into SFP "A". This represents the limiting or shortest time for a pool to boil.
A & B (End of cycle):	This represents a case which involves the condition at the end of a fuel cycle after a full core off load has decayed. This condition is less limiting than the BOC case.
C & D (1.0 MBTU/Hr):	This case represents a situation in which only a small amount of 5 year old fuel[1] is placed in the C pool.
C & D (15.6 MBTU/Hr):	This case represents a situation in which the C & D pools are filled with spent fuel, all of which is 5 years or older.

A.4 NORMAL WATER MAKEUP TO FUEL POOLS

Multiple water makeup sources to the A & B SFPs are available and proceduralized. This section discusses these proceduralized makeup methods, and Section A.5 discusses some non-proceduralized methods. Following the installation of plant modifications associated with SFPs C and D, a completely redundant SFP cooling system, purification system, and skimmer system will be installed in the North end of the FHB. This will provide redundant delivery locations for operators to align existing makeup water sources to SFPs C and D, transfer canals, and the cask loading pool. Operating procedures (OP-116) will be revised to reflect the redundant makeup water pathways to SFPs C and D prior to adding spent fuel to pool C.

Normal makeup to the pools and canals is accomplished by aligning the purification pumps to take suction from the demineralized water (DW) system. This is done by either opening locked closed manual valve 1SF-201 or 2SF-201 with the FPCCS Cleanup/Purification Subsystem in operation. These valves are located in the South and

[1] Fuel that has been removed from the RPV for more than 5 years.

North ends of the 216 ft Elevation of the FHB, respectively. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.4.

CP&L [A-1] identified that the purification pumps are not required to run for success of this path. Demineralized water system pump operation is likely required. The flow paths for use of DW into the SFPs includes this method without the purification pumps running. Therefore, while the preferred and normal method of makeup is through the purification system pumps, the purification pumps need not to be running to obtain flow into the SFP through the normally open suction line up[1]. [Eric McCartney, 9/29/00]. The source of water is the demineralized water storage tank, which has a capacity of 500,000 gallons. The flow rate is 100 gallons per minute. The operator can initiate this flow path in approximately five minutes, excluding any transit time.

Table A-1 is a summary of the normal and supplemental SFP makeup methods (See Section A.5 for discussion of the supplemental makeup methods). Table A-1 identifies the normal methods of SFP makeup to be from the DW system to the SFP via the locked closed manual valves on the 216' elevation of the FHB. This is labeled as method PB in Table A-1.

(1) Because the purification system is normally operating, the manual suction valves are open to at least one of the SFPs associated with the system. This is estimated at 99% by CP&L [Eric McCartney, 9/29/00]

In the following figures, the valve positions under normal operation are shown. The following indicates valve position:

- "Blackened" valve – normally closed
- "White" valve – normally open

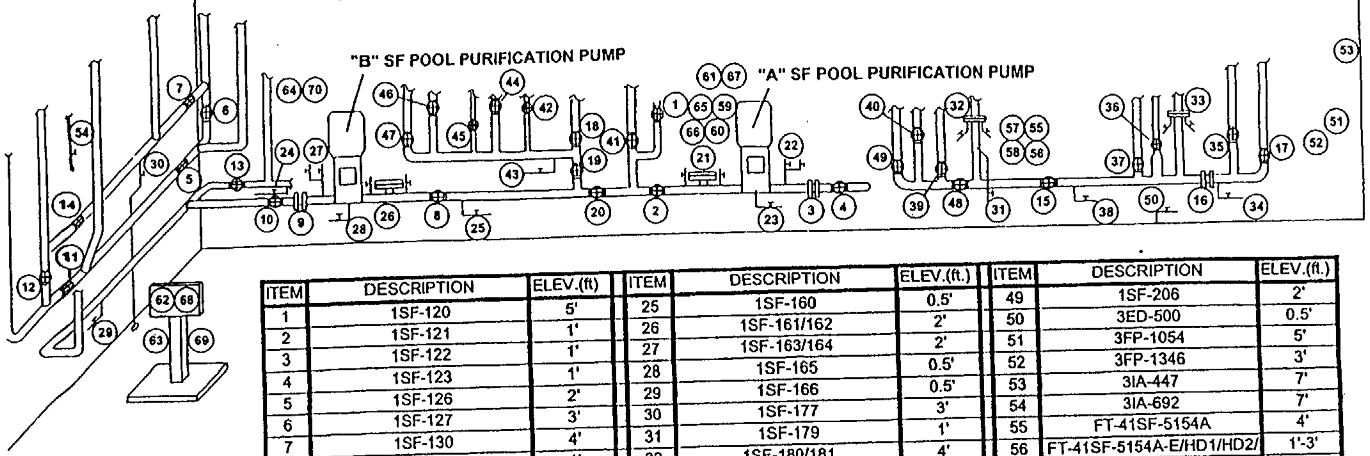
Figure A.4-2 shows the FHB South 216' Elevation and the specific locked closed manual valve that needs to be opened (1SF-201). This arrangement is similar to that in the North 216' Elevation.

Figure A.4-3 shows a simplified diagram of the flow path through 2FS201 and back through the suction of the clean up pumps.

Figure A.4-3 is similar to Figure A.4-2 except it shows the pathway into SFPs C&D through FPCCS when clean up is not in service. Manual valve 2SF-201 is required to be opened.

Figures A.4-4, A.4-5, and A.4-6 are simplified schematics for pathways when the FPCCS cleanup or skimmer pump is in service. These pathways are beneficial under most non-severe accident conditions. However, for the Postulated Sequence included in the ASLB Order, these line ups are not substantial benefits.

FHB 216' SOUTH - SPENT FUEL POOL PURIFICATION



ITEM	DESCRIPTION	ELEV.(ft)
1	1SF-120	5'
2	1SF-121	1'
3	1SF-122	1'
4	1SF-123	1'
5	1SF-126	2'
6	1SF-127	3'
7	1SF-130	4'
8	1SF-131	1'
9	1SF-132	1'
10	1SF-133	1'
11	1SF-136	2'
12	1SF-137	3'
13	1SF-138	1'
14	1SF-139	4'
15	1SF-141	1'
16	1SF-142	1'
17	1SF-143	2'
18	1SF-148	3'
19	1SF-149	2'
20	1SF-150	1'
21	1SF-151/152	2'
22	1SF-153/154	2'
23	1SF-155	0.5'
24	1SF-156	0.5'

ITEM	DESCRIPTION	ELEV.(ft.)
25	1SF-160	0.5'
26	1SF-161/162	2'
27	1SF-163/164	2'
28	1SF-165	0.5'
29	1SF-166	0.5'
30	1SF-177	3'
31	1SF-179	1'
32	1SF-180/181	4'
33	1SF-182/183	4'
34	1SF-184	0.5'
35	1SF-187	3'
36	1SF-188	3'
37	1SF-189	2'
38	1SF-190	0.5'
39	1SF-191	2'
40	1SF-192	3'
41	1SF-193	4'
42	1SF-194	4'
43	1SF-195	2'
44	1SF-200	4'
45	1SF-201	3'
46	1SF-202	4'
47	1SF-203	3'
48	1SF-205	1'

ITEM	DESCRIPTION	ELEV.(ft.)
49	1SF-206	2'
50	3ED-500	0.5'
51	3FP-1054	5'
52	3FP-1346	3'
53	3IA-447	7'
54	3IA-692	7'
55	FT-41SF-5154A	4'
56	FT-41SF-5154A-E/HD1/HD2/ HI1/HV1/LD1/LD2/LI1/LV1	1'-3'
57	FT-41SF-5154B	4'
58	FT-41SF-5154B-E/HD1/HD2/ HI1/HV1/LD1/LD2/LI1/LV1	1'-3'
59	PI-41SF-5190A	5'
60	PI-41SF-5190A-D1/D2/I1/V2	0.5'-4'
61	PI-41SF-5190A-V1	5'
62	PI-41SF-5190B	5'
63	PI-41SF-5190B-D1/D2/I1/V2	0.5'-3'
64	PI-41SF-5190B-V1	5'
65	PS-41SF-5190A	5'
66	PS-41SF-5190A-D1/D2/I1/V2	0.5'-4'
67	PS-41SF-5190A-V1	5'
68	PS-41SF-5190B	5'
69	PS-41SF-5190B-D1/D2/I1/V2	0.5'-3'
70	PS-41SF-5190B-V1	5'

FHB 216' South - Spent Fuel Pool Purification

Figure A.4-1 FHB 216' South - Spent Fuel Pool Purification

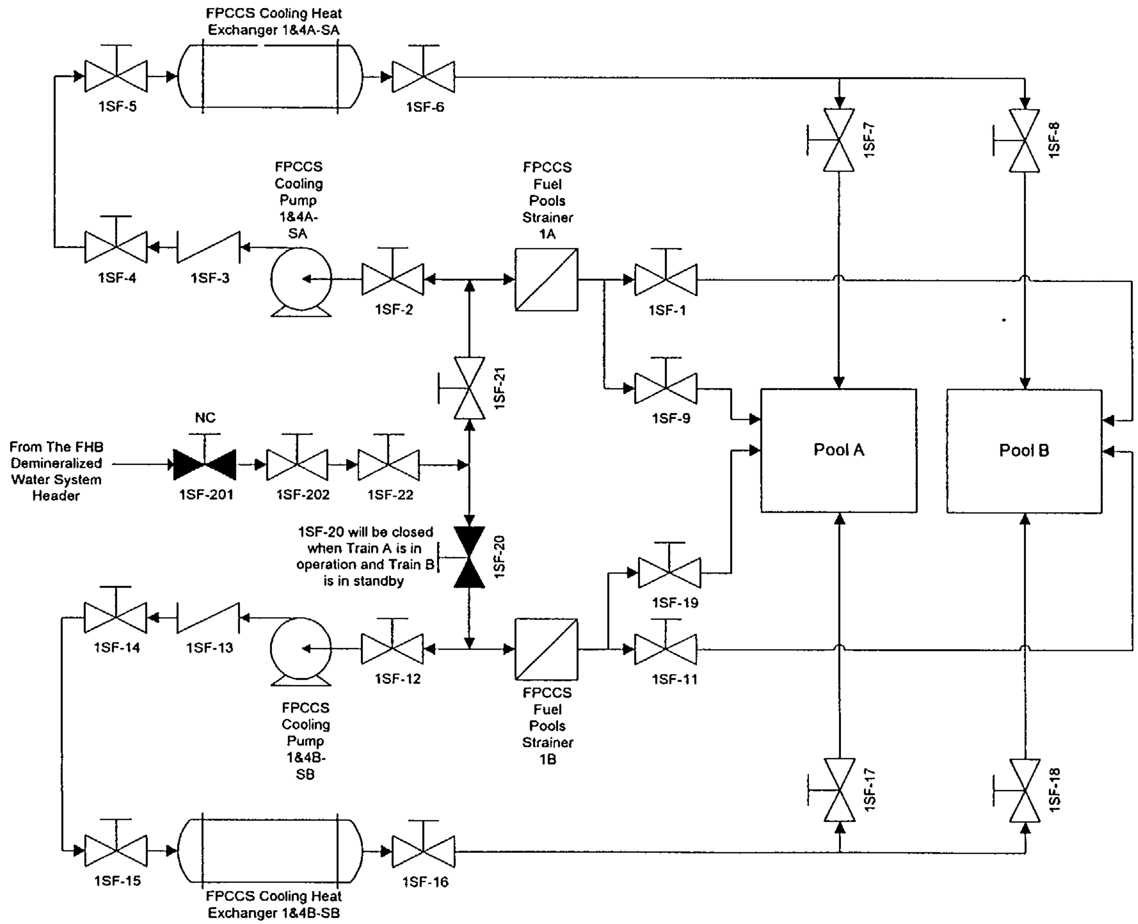


Figure A.4-2 Demineralized Water Makeup to Pools A and B with FPCCS Cleanup Not in Service

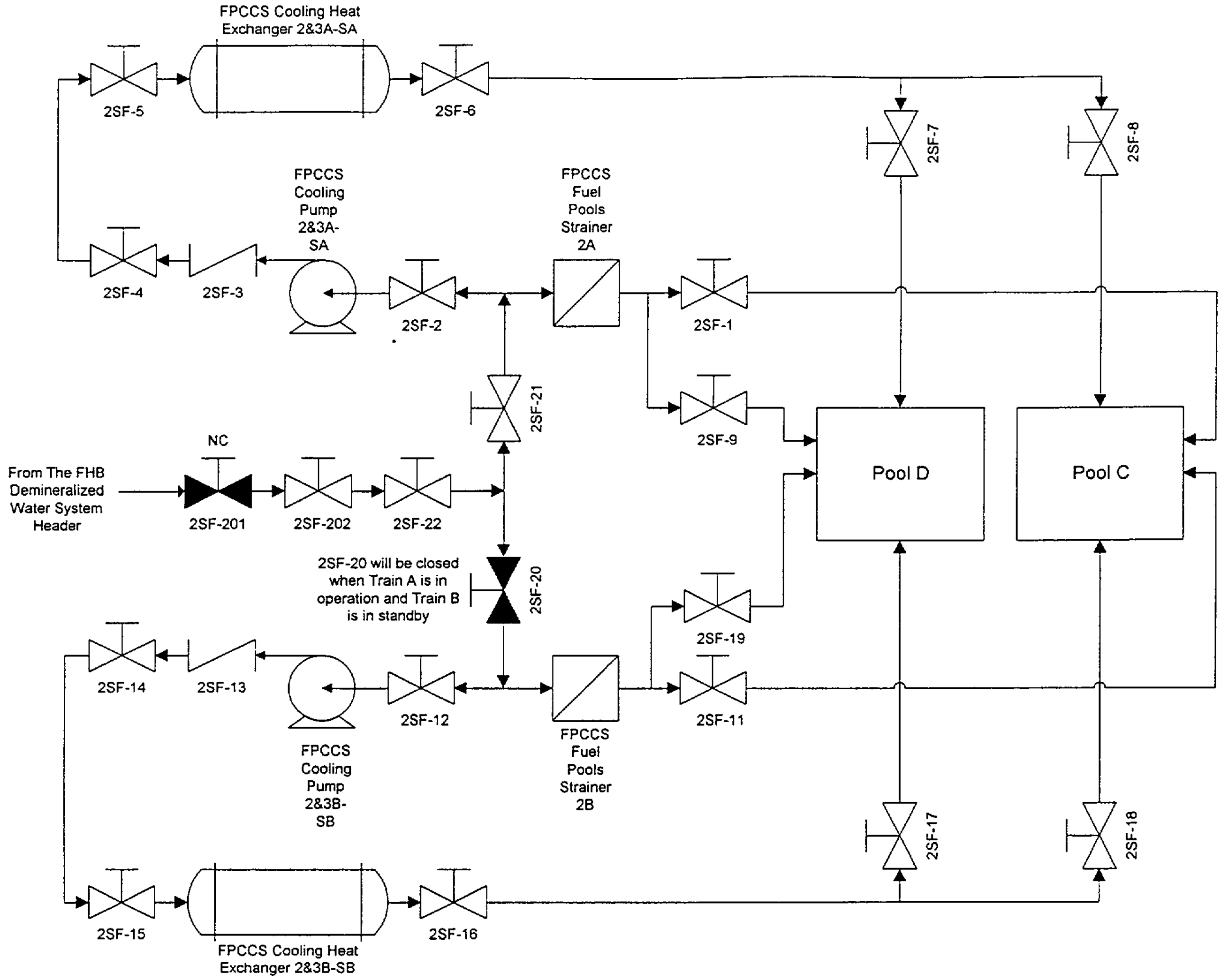


Figure A.4-3 Demineralized Water Makeup to Pools C and D with FPCCS Cleanup Not in Service

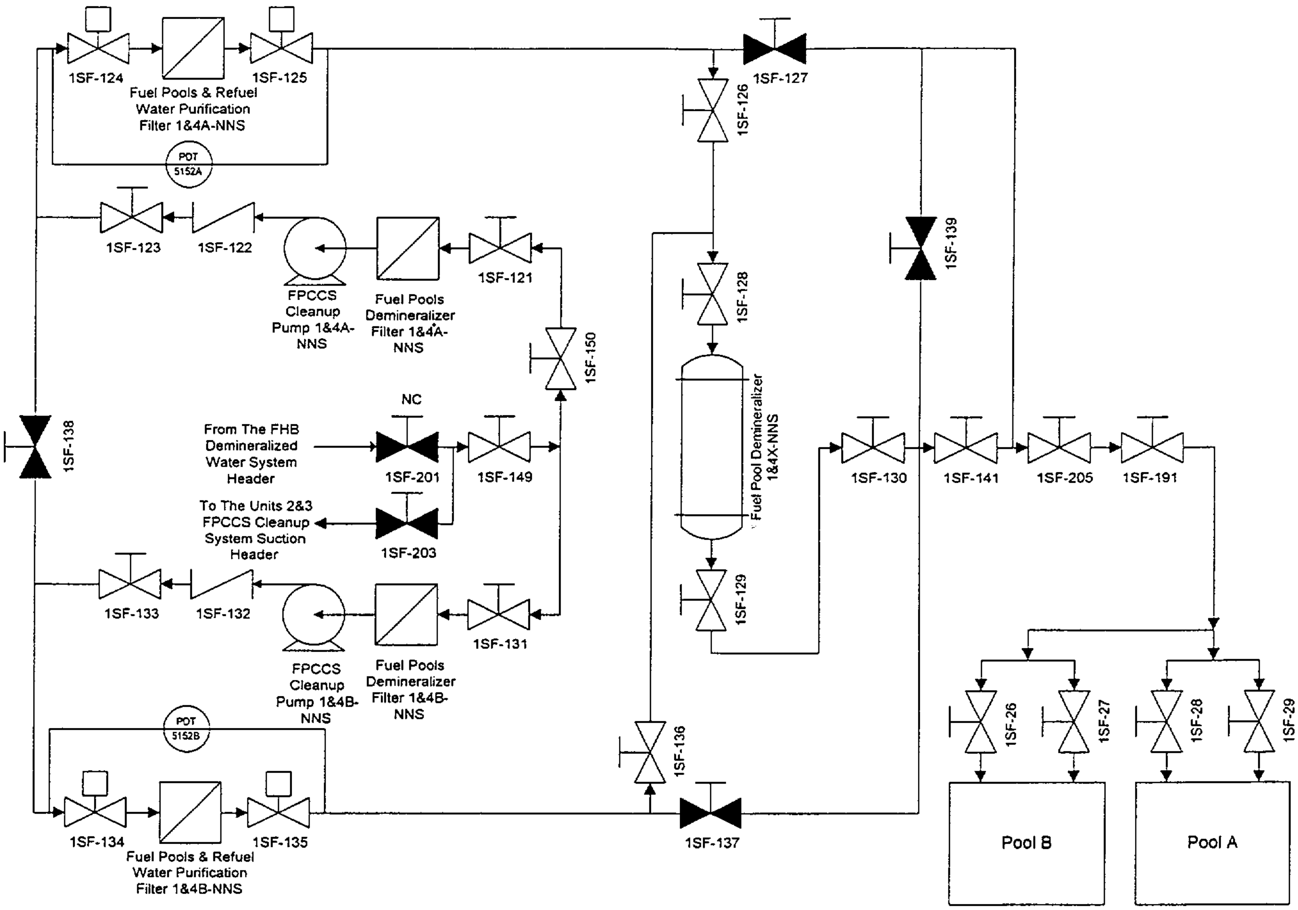


Figure A.4-4 Demineralized Water Makeup to Pools A and B with FPCCS Cleanup in Service

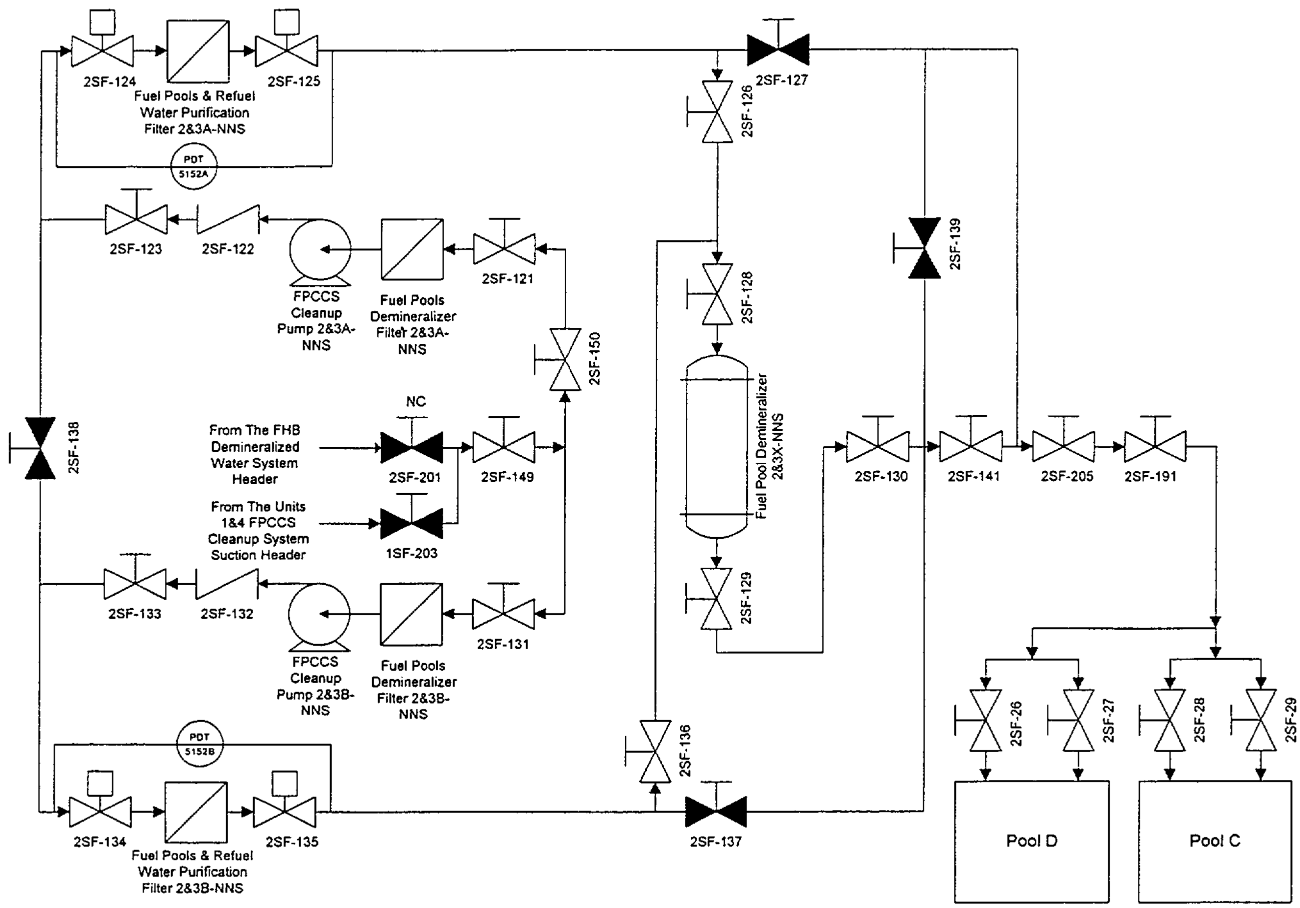


Figure A.4-5 Demineralized Water Makeup to Pools C and D with FPCCS Cleanup in Service

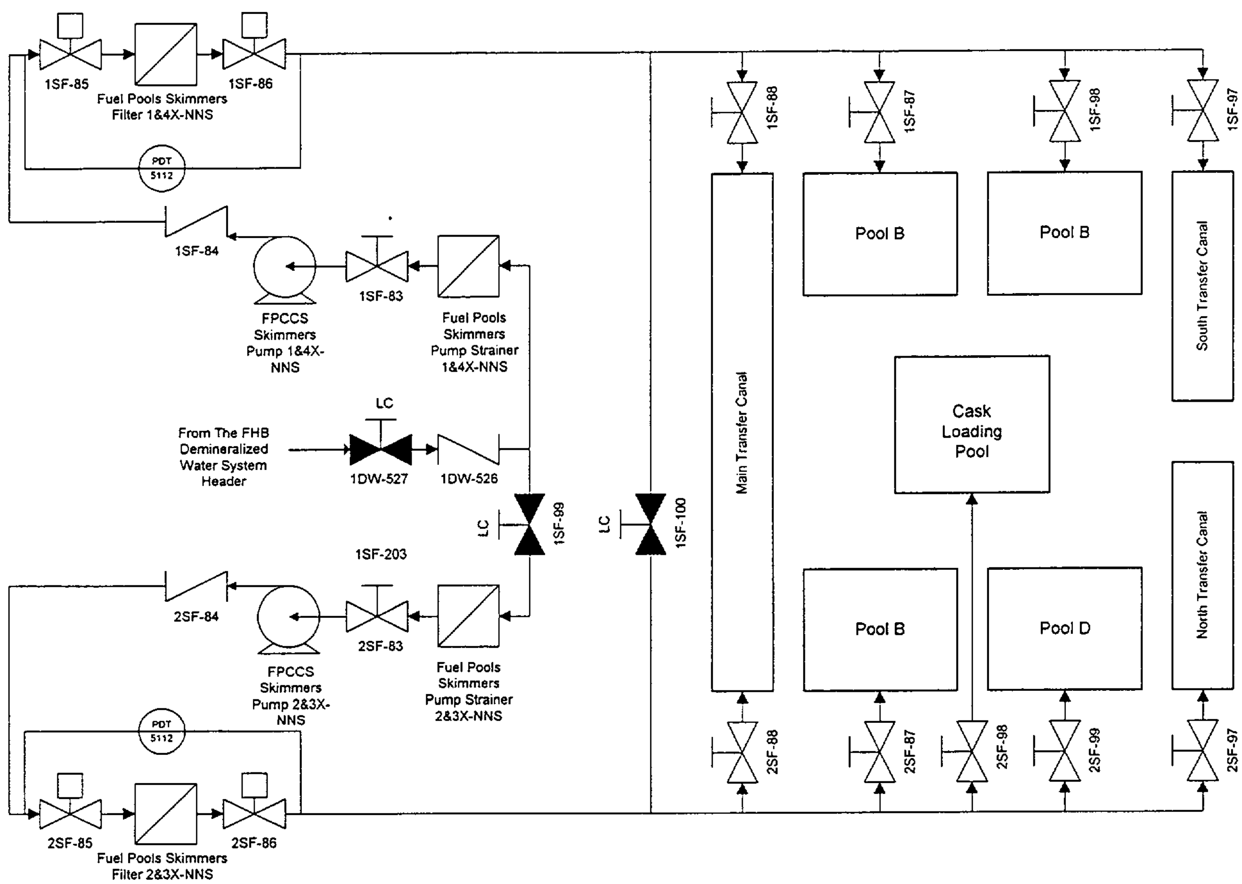


Figure A.4-6 Demineralized Water Makeup to Pools and Canals with FPCCS Skimmers in Service

A.5 SUPPLEMENTAL WATER MAKEUP TO FUEL POOLS

In the event of a loss of SFP water inventory, SFP low level alarms would be received in the Main Control Room at Auxiliary Equipment Panel Number 1. SHNPP annunciator panel procedure APP-ALB-023, *Auxiliary Equipment Panel No. 1*, directs the operators to initiate makeup to the SFPs per Plant Operating Manual Operating Procedure OP-116, *Fuel Pool Cooling and Cleanup*. Table A-1 summarizes the supplemental SFP makeup methods. These methods include both proceduralized and non-proceduralized methods. In the event that normal makeup from the demineralized water system through the FPCCS Cleanup / Purification Subsystem is not available, OP-116 gives the options provided in Table A-1.

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
Proceduralized Methods									
PA.	ESW (Alt. #5)[1]	OPP-116 (8.13)	30 min.[2] to 1 hr	FHB[3] 236' El. RAB 236' El.	ESW and ESW Booster	Div. I or II	Uniform Hazard Response System upper or lower reservoir	50 - 75 gpm	Large

(1) The alternate number references are those provided in the first interrogatory response to NRC issued September 26, 2000 regarding the ASLB order.

(2) Need to also have complement of people.

(3) Not required.

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
PB*	Demin Water (Normal Makeup)	OPP-116 (8.4)	~ 30 min. (5 min. Excluding Transit Time)	FHB 216' El. North(4) or South(5) Valves 1SF 201 South(5) 2SF 201 North(4)	• Demin Pumps • Cleanup Pumps are part of procedure but not required(6)	Offsite Power(7) (AOVs not required)	Demin water tank	100 gpm with Demin pumps only (2" pipe)	500,000

* Normal Makeup Supply

(4) Makeup flow would be directed to the C & D Pools.

(5) Makeup flow would be directed to the A & B Pools.

(6) The normal operating range for the demin water system header pressure is 150 psig to 225 psig. Therefore, a minimum supplied head through 2SF-201 would conservatively be 100 psig (assuming a 50 # headloss through the piping) which would result in at least 100 gallons per minute. The status of the purification pump would have little or no impact on the delivery flow rate of demin water to the system. (Personal Communication Eric McCartney (CP&L) to E.T. Burns (ERIN), October 4, 2000)

(7) Emergency supply would require ad hoc alignment.

Table A-1

SPENT FUEL POOL MAKEUP METHODS

	Method	Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
PC	RWST (Alt #2)	OPP-116 (8.5)	30 min.	• FHB 216 ft. El. valve 1SF-193; and, • RAB 236 ft. El. Valve 1CT-23 • FHB 236 ft. El. or FHB 286 ft. El. for pump breaker	• N/A through suction path; or, • FPCCS Cleanup pumps through discharge path	N/A	RWST (Gravity Drain)	100 gpm	• 490,000 • May be unavailable because already discharged to containment
PD	RWMST (Alt #6)	OPP-116 (8.26)	30 min.	• RAB 236' • FHB 236'	Rx water M/U pumps or Gravity feed is feasible under certain conditions	Div. I & II	RWMST	75 - 100 gpm	80,000 (usually full)
PE	Demin to Fuel Pool Skimmer (Alt #3)	OPP-116 (8.6)	60 min. (Est.)	FHB 236' El. 1 valve	• Demin pumps • Skimmer pumps	Offsite Power	Demin water tank	100 gpm	500,000

Table A-1

SPENT FUEL POOL MAKEUP METHODS

	Method	Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
PF	RWST to FPCC CLG pumps (Alt #4)	OPP-116 (8.12)	30 min.	FHB El. 236' FHP El. 216' RAB El. 236'	Gravity drain is adequate	• None for gravity drain • Div. I or II for pump operation	RWST	• 60 - 100 gpm by gravity • 5000 gpm with FPCC cooling pump operating	• 490,000 • May already be discharged to containment
PG	Demin Water to FPCC cleanup system (Alt #1)	OPP-116 (8.5)	30 min.	FHB El. 236' FHB El. 216' FHB El. 261' El. for pump breaker'	Cleanup pump	Offsite Power	Demin Water Tank	100 gpm with cleanup pumps running	500,000
PH	RWDT	OPP-116 (8.22)	More than 30 min.	FHB	Not Evaluated	Not Evaluated	RWDT during normal operation	Not estimated	Water not likely available during accident conditions
Non Proceduralized Methods									
N1	Fire Protection to hoses on 286' El. of FHB	None	30 min.	FHB 286' El.	Diesel Fire Pump or Electric Fire Pump	None	Upper Lake only (seismic guaranteed source)	~ 100 gpm per hose	Large

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
N2	Demin Water Quick Connect Options on 286' El.	None	30 min.	286' El. FHB	Demin Water	Offsite	Demin Water Tank	100 gpm (2" pipe)	500,000
N3	NSW	None[4]	> 60 min.	WPB	NSW	Offsite	Lake	> 100 gpm	Large

[4] 300 ft of hose would be required. This is currently not prestaged.

1. Emergency Service Water (ESW) System – The ESW system may be connected to dedicated FPCCS Cooling Subsystem emergency makeup connection vent valve 1SF-76 (located downstream of 1CT-23 at the 236 ft elevation of the RAB, column line E42 above the heat exchanger valve gallery) via approximately 50 feet of 1 inch rubber hose. This hose is stored in a gang box located in the stairwell opposite 1CT-23 (through door D605) at the 236 ft elevation of the RAB. The ESW valves are located in the overhead in the hallway just outside the hot machine shop (1SW-1239 for ESW train B) and in the overhead just inside the hot machine shop (1SW-269 for ESW train A) in the RAB at the 236 ft elevation, column line D43. The source of water is the Harris Lake, which provides a virtually unlimited source of water. The flow rate is approximately 50 to 75 gallons per minute. The operator can align this flow path within 30 minutes. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.13. (Table A-1, Method PA)

2. RWST – Normally closed manual valves 1SF-193, located in the FHB at the 216 ft elevation (north) and 1CT-23, located in the RAB at the 236 ft elevation, column line E13 must be opened to align the FPCCS Cleanup / Purification Subsystem to the RWST. After aligning the valves, the operator turns on power supply breakers for the purification pumps and starts the pump from one of two locations, the 236-foot elevation FHB or the operating deck of the FHB. The source of this flow path is the RWST with a capacity of 490,000 gallons. The flow rate is 100 gallons per minute. The operator can align this flow path within 30 minutes. If the RWST is full, this flow path will result in gravity flow to the spent fuel pools, transfer canal, or cask loading pool without needing any pumps due the elevation difference between the RWST and the spent fuel pools. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.5. (Table A-1, Method PC)

   The RWST is not filled during refuel operations with the cavity flooded; therefore, use of the RWST as a makeup water source to the SFP is precluded under those conditions. In addition, the RWST can be used for injection to containment during a severe accident, therefore it is likely not available for SFP makeup under the conditions postulated in the ASLB Order.

3. Primary Makeup Water System (PMWS) – Locked closed manual valve 7PM-V238-1 provides isolation between the FPCCS and the PMWS. This valve is located in the RAB on the 236 ft elevation. Opening this valve and aligning four manual valves in the FHB equipment room at the

236 ft elevation allows water from the 80,000 gallon Reactor Makeup Water Storage Tank (RMWST) to be used to fill the FHB pools and canals. The source of water is the RMWST with a capacity of 80,000 gallons. The flow rate is 75 to 100 gallons per minute. The operator can align this flow path within 30 minutes. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.26. (Table A-1, Method PD)

4. Demineralized Water (DW) System – Normally locked closed manual valve 1DW-527, located in the FHB equipment room at the 236 ft elevation, may be opened when the FPCCS Skimmer is in service to slowly add DW to the pools and canals through their floating skimmers. The source of water is the demineralized water storage tank with a capacity of 500,000 gallons. The flow rate is approximately 100 gallons per minute. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.6. (Table A-1, Method PE)

5. RWST to FPCC Cooling Pumps – To align the RWST to the suction of the FPCCS Cooling Subsystem pumps the operators must align eleven manual valves. This will deliver water to the South Transfer Canal, the Main Transfer Canal and the Cask Loading Pool. Eight of these valves are in the FHB equipment room at the 236 ft elevation, two valves are in the south end room of the FHB at the 216 ft elevation and 1CT-23 is located in the RAB at the 236 ft elevation, column line E13. If the RWST level is high, then the transfer canal or cask loading pool will fill due to gravity. The SFP cooling pump is then started from the Main Control Room. The source of water is the RWST with a capacity of 490,000 gallons. The flow rate is 5000 gallons per minute. The operator can align this flow path within 30 minutes. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.12. (Table A-1, Method PF)

6. Demineralized Water System – To makeup water to SFPs "A" and / or "B," the operators must align four manual valves. (See OP 116 Section 8.5). Two are located in the FHB equipment room at the 236 ft elevation and two are located in the south end room at the FHB 216 ft elevation. To makeup water to SFPs "C" and / or "D," the operators must align two manual valves in the FHB equipment room at the 236 ft elevation and two additional manual valves located in the north end room at the FHB 216 ft elevation. Once the power supply is turned on, the operator turns on the purification pump at one of two locations, the operating deck of the FHB or the 236-foot elevation of the FHB. The source of water is the demineralized water storage tank with a capacity of 500,000 gallons.

The flow rate is 100 gallons per minute. The operator can initiate flow in approximately 30 minutes, excluding any transit time. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.5. (Table A-1, Method PG)

7. RWDT – This method is considered viable during nominal operation for small quantities of makeup. It is not credited for larger volume during accidents. (Table A-1, Method PH)

There are several other potential sources of makeup to the SFPs that are not currently credited in SHNPP Operating Procedure OP-116. These non-procedural lineups may be attempted under the direction of the SHNPP Technical Support Center (TSC):

1. Fire System – The FHB is equipped with a fire header that runs along the east and west walls on the 286 ft elevation. There are three hose stations (each containing a 1.5" hose) along the west wall and four hose stations along the east wall on the 286 ft elevation operating floor connected to this header. Any or all of these hoses could be directed into the pools the canals to supply more than 100 gpm per hose. The fire protection system draws water from upper Harris Lake via a motor driven fire pump or a redundant diesel driven fire pump. (Table A-1, Method N1)

   It is noted that the Fire Protection System capability to provide SFP makeup may become more complicated under a seismic event. A seismic event may lead to the failure of the fire protection pumps (i.e., they are not seismic). However, the piping is seismic. The SHNPP method of supplying fire protection water is through the use of the ESW pumps, which are seismically qualified, through 2 manual cross connect valves located on 236' El. of RAB.

2. Demineralized Water (DM) System – There are 19 DM stations located along the east and south walls of the FHB operating deck at the 286 ft elevation. Each of these stations has a manual isolation valve and a standard quick disconnect fitting. Rubber hoses with matching fittings are readily available on the FHB operating deck at all times for routine work. Hoses could be quickly attached to any or all of these DM stations and directed into any of the pools and / or canals. (Table A-1, Method N2)

3. Normal Service Water (NSW) System – The NSW System extends into the Waste Processing Building (WPB) at the 261 ft elevation near the WPB stairwell that leads up to the south end of FHB 286 ft elevation. Approximately 300 feet of 1 inch rubber hose could be connected to any one of a number of 1 inch drain valves on the NSW lines in this area, run up the stairwell and directed into pool "A". (Table A-1, Method N3)

A.6 FUEL POOL INSTRUMENTATION

The critical levels in the SFPs are summarized in the following table:

Top of Pools/Canals	286.000 feet
Top of an installed gate	285.125 feet
HI Level Alarm in Main Control Room	284.900 feet
Normal water level	284.500 feet
LO Level Alarm in Main Control Room	284.000 feet
Technical Specification 3.9.11 Limit	283.790 feet
LO-LO Level Alarm in Main Control Room	282.000 feet
Top of BWR racks in Pools "B", "C" & "D"	261.250 feet
Top of PWR racks in Pools "B", "C" & "D"	260.480 feet
Top of PWR racks in Pool "A"	260.960 feet
Bottom of Main Transfer Canal	260.000 feet
Bottom of North / South Transfer Canals	251.000 feet
Bottom of fuel pools	246.000 feet
Bottom of Cask Loading Pool	240.000 feet

Monitoring capability of the SFPs at SHNPP can be summarized in the following table:

Monitoring Capability	Spent Fuel Pools			
	A	B	C[4]	D[4]
• Camera • Pool Level Indicator • Pool Level Alarm	None No Yes[2]	None No Yes[2]	None No Yes[2]	None No Yes[2]
• FPCCW Pump Flow (Lose Suction at -4 ft.)	No[1], [3]	No[1], [3]	No[1], [3]	No[1], [3]
• Temperature Alarm - Bistable Hi Level, - Lo Level - Lo-Lo Level	Control Room Indication	Control Room Indication	Control Room Indication	Control Room Indication
• Local Indications Level	Observation	Observation	Observation	Observation
• Radiation (.1 mr/hr - 10^3 mr/hr)	Local at 286' El. FHB	Local at 286' El. FHB	Local at 286' El. FHB	Local at 286' El. FHB

[1] Local flow and pressure drop indications in FHB are available

[2] 22 ft. above fuel

[3] Lose temperature and suction

[4] Equivalent instrumentation is projected to be available following activation of Pools C & D

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[A-2] CP&L Drawing, Fuel Handling Building Bulkhead Gates & Details - Units 1 & 2, CAR-2168-G-125 Revision 4

[A-3] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 6, Annunciator Panel Procedure, Auxiliary Equipment Panel No. 1, APP-ALB-023, Revision 19

[A-5] Carolina Power and Light Company (CP&L), Shearon Harris Nuclear Power Plant (SHNPP), Plant Operating Manual, Volume 6, Part 2, System Description, Fuel Pool Cooling and Cleanup System, SD-116, Revision 7

[A-6] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 2, Operating Procedure, Fuel Pool Cooling and Cleanup, op-116, Revision 117

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[A-8] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 6, Annunciator Panel Procedure, Local Control Panel F-P7, APP-F-P7, Revision 4

[A-9] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 6, Annunciator Panel Procedure, Local Control Panel F-P9, APP-F-P9, Revision 4

[A-10] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Fuel Pools Cooling System - Unit 1 (CPL-2165-S-0805), Page 6.1.1, Revision 0

[A-11] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Fuel Pools Cooling System - Unit 2 (CPL-2165-S-0807), Page 6.1.3, Revision 0

[A-12] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Fuel Pools Clean-Up Systems - Sheet 1 - Unit 1 (CPL-2165-S-0561), Page 6.1.5, Revision 0

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[A-15] CP&L, SHNPP Drawing, Plot Plan, CAR-2165-G-002, Revision 20

[A-16] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 190.00' & 216.00', CAR-2165-G-015, Revision 16

[A-17] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 236.00', CAR-2165-G-016, Revision 20

[A-18] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 261.00', CAR-2165-G-017, Revision 19

[A-19] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 286.00', CAR-2165-G-018, Revision 20

[A-20] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 305.00', CAR-2165-G-019, Revision 20

[A-21] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Section Sheet 1, CAR-2165-G-020, Revision 19

[A-22] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Sections Sheet 2, CAR-2165-G-022, Revision 21

[A-23] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Plans - Sheet 1, CAR-2165-G-022, Revision 19

[A-24] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Plans - Sheet 2, CAR-2165-G-023, Revision 10

[A-25] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Sections - Sheet 1, CAR-2165-G-024, Revision 10

[A-26] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Sections - Sheet 2, CAR-2165-G-025, Revision 15

REFERENCES (Cont'd)

[A-27] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Sections - Sheet 3, CAR-2165-G-026, Revision 13

[A-28] Interrogatory update from CP&L. Also discussed in Steven Edwards Personal Communication to E.T. Burns (ERIN) on October 30, 2000.

Appendix B
DISCUSSION OF REMOTE AND SPECULATIVE

B.1 PURPOSE

The purpose of this appendix is to offer an estimate of the frequency of events that may be considered in a category of "remote and speculative" such that the risks associated with the event are generally considered acceptable by society at large, notwithstanding the consequences.

B.2 DISCUSSION

In all human endeavors, it is prudent to plan for those natural and man-made occurrences that can be reasonably foreseen. However, not every imagined, rare event can be explicitly evaluated and analyzed in every detail.

All citizens struggle with the balancing of "acceptable" risks associated with occupation, lifestyle, and environmental factors. Not only do these three choices impose a broad spectrum of potential risks, but they also are perceived differently by individuals. Nevertheless, society (the group of individuals) must form a consensus on these risk perceptions when the choices cross into the realm affecting society at-large.

Using risk assessment techniques and learning to perceive risks correctly is an important goal for the nuclear industry. As a society, decisions on where to expend time, energy, and resources of both people and money need to be made. Although it is sometimes convenient and satisfying to concentrate on a very minor problem and ignore the larger ones, society must make difficult decisions regarding the allocation of scarce resources. Therefore, it is useful to formulate a criteria to judge when an event is of such low frequency as to be inconsequentially small irrespective of the consequences.

B.3 PROPOSED CRITERIA

A consensus has developed within the nuclear industry and within the NRC regarding the “de minimus” point, that is, a frequency of events where the substantial uncertainties of nature and life create a point at which the risk becomes sufficiently small as to seriously question whether the frequency of such events can be effectively reduced below this level. This point has generally been placed in the frequency range of less than 1 in a million per year (i.e., 1E-6/yr). Reference can be made to the large body of work that the NRC has compiled related to:

- The Severe Accident Policy Statement
- Safety Goals for the Operation of Nuclear Power Plants; Policy Statement. [51FR 280444, dated 8/4/86; 51FR 30028, dated 8/4/86] and SECY-91-270.
- The NRC Backfit Policy 10 CFR 50.109

These documents and their supporting analyses indicate the following:

- This Safety Goal Policy statement focuses on the risks to the public from nuclear power plant operation. Its objective is to establish goals that broadly define an acceptable level of radiological risk.
- Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.

In each of these documents, the most severe accidents (large release to the public) are expected to be limited in frequency to below 1E-6/reactor year. This can be interpreted to be the level at which risk is sufficiently low that further attempts to reduce the risk level become difficult or impossible to justify. (See NRC Backfit Policy 10CFR50.109.)

In addition to past NRC work in this area, frequencies of events that are based on historical records are shown in Figure B-1 for reference. The "de minimus" point, or the point at which events may be so remote and speculative as to be below what can be rationally considered, is also indicated at 1E-6/yr. For the purposes here, events with frequencies below this "de minimus" point can be referred to as "remote and speculative".

Risk reduction below the "de minimus" point might be accomplished by eliminating a product or service; however, in most cases society has decided that this is not suitable because it interferes with individual freedom and may in fact introduce new or competing risks that may be larger than the risks being "eliminated."

B.4 CONCLUSION

Events with frequencies below one in a million per year (1E-6/year) can be considered to be sufficiently low in frequency such that additional efforts by society to reduce the frequencies below this level are not considered warranted.

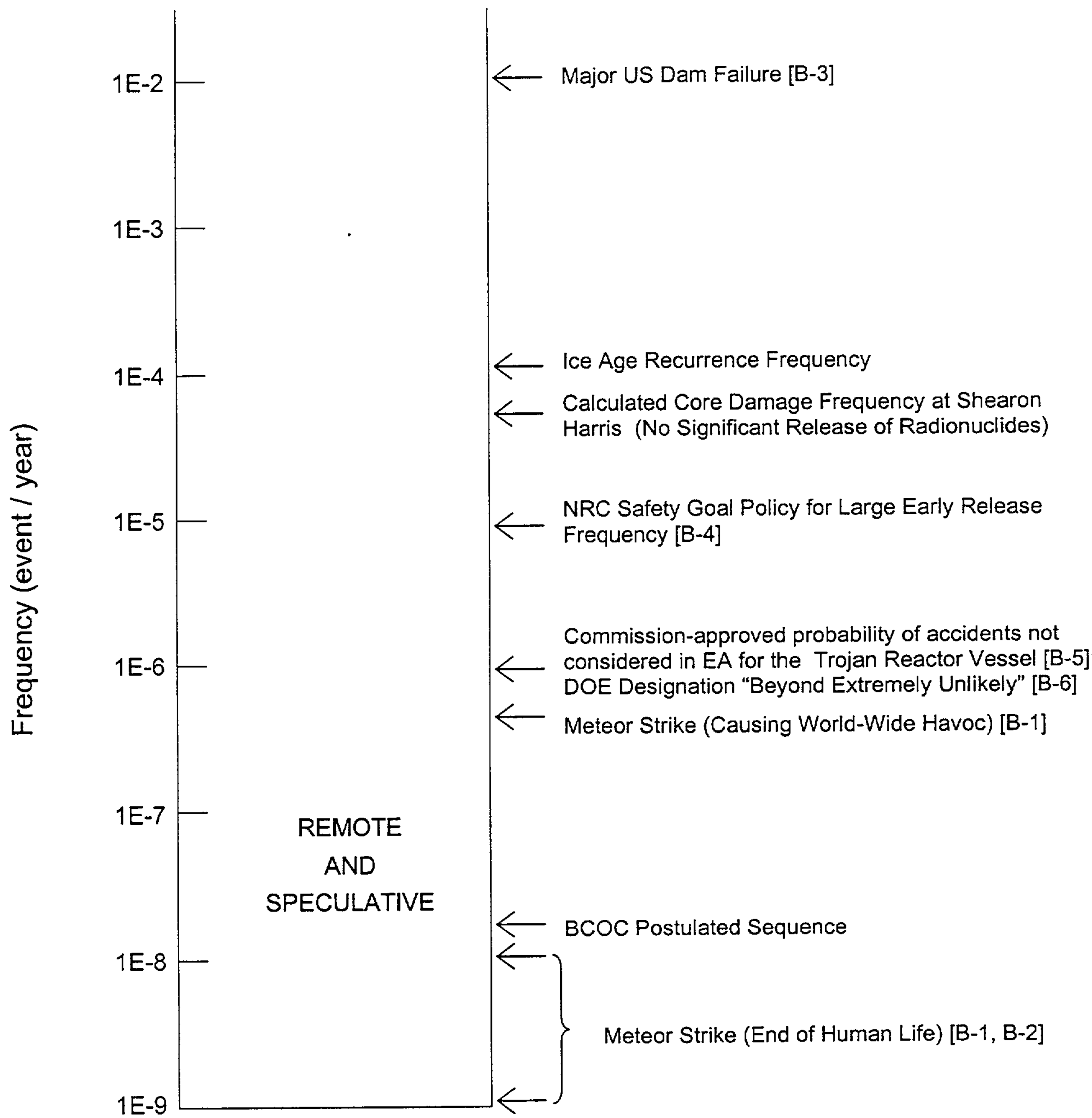

Figure B-1 Comparative Insights into "Remote and Speculative" Events

REFERENCES

[B-1] The Safeguard Survey, NASA-TM-107979, dated January 25, 1992.

[B-2] Evaluation of External Hazards to Nuclear Power Plants in the United States, NUREG/CR-5042, Supplement 2, dated February 1989.

[B-3] "Disasters as a Necessary Part of Benefit - Cost Analyses", Science, Vol. 197, September 1977.

[B-4] SRM on SECY-00-0077, "Modifications to the Reactor Safety Goal Policy Statement" (June 27, 2000).

[B-5] SECY-98-231, "Authorization of the Trojan Reactor Vessel Package for One-Time Shipment for Disposal" (October 2, 1998).

[B-6] DOE-STD-3009-94, "Preparation Guide for U.S. Department of Energy Non Reactor Nuclear Facility Safety Analysis Reports," Change Notice No. 1 (January 2000).

Appendix C
HUMAN RELIABILITY ANALYSIS

C.1 INTRODUCTION

This appendix is a summary of the critical aspects of the Human Reliability Analysis (HRA) performed to support the PSA to address the Postulated Sequence.

C.2 METHODOLOGY

The Human Reliability Analysis (HRA) to support the evaluation of operator actions in the PSA of the Postulated Sequence is a combination of methods that have been used successfully in past nuclear power plant PSAs, both operating and shutdown. These methods address both short duration responses which may be time critical and very long duration responses that may be strongly dependent on other performance shaping factors such as local access.

The model is structured to ask multiple questions regarding the operator action success:

- How is the action diagnosed and by whom?

  This is answered by identifying a common basic event for all makeup sources that requires the TSC to diagnose the action and direct the proper response.

- How is the action carried out?

  This is represented by an assessment of the manipulation error using the cause based method [C-2] supplemented by ASEP [C-3], as appropriate.

- How does accessibility play a role?

  Accessibility is treated separately from the above diagnosis and execution evaluations. The deterministic MAAP calculations assess whether the conditions in the local areas are adequate to allow the local manual actions. If so, then the manipulation error determined above applies; if not then the action is considered to have failed.

- What are the critical performance-shaping factors?

  The ability to accurately characterize the HEP is contingent upon identifying the interplay among the performance-shaping factors that influence the response. These include: time available, time required, competing tasks, degree of complexity, lighting, accessibility (see previous item), threat to health and safety.

The following discussion provides an overview of the dominant contributors involved in the determination of the HEPs.

The approach for the Human Error Probability (HEP) evaluations places heavy emphasis on the accident sequence conditions imposed on the operating crew. The sequence definition determines performance-shaping factors such as: accessibility, time available, and degree of threat to health and safety. Therefore, while the operating crew actions are the same to successfully align a system, the cues and the imposed severe accident conditions may create substantially different estimates of successfully completing an action.

For situations with no or limited adverse conditions outside containment, all methods are viable options for protecting the SFP if support systems have not failed as part of the accident sequence cutsets.

For situations where only the Reactor Auxiliary Building (RAB) has encountered adverse environmental conditions, the options remaining include:

Proceduralized

PB -- Demineralized Water to FPCC (OP 116, 8.4)

PE -- Demineralized Water to FPCC Skimmer Pumps (OP 116, 8.6)

PG -- Demineralized Water to FPCC Cleanup System (OP 116, 8.5)

PH -- RWDT -- The limited RWDT water supply may lead to the need for supplementary makeup

Non-Proceduralized

N1 -- Fire Protection to hoses on Refuel Floor

N2 -- Demineralized Water to Quick Disconnect Fittings

For situations where substantially degraded conditions exist in the RAB and those conditions have influenced the Fuel Handling Building (FHB) also, the options remaining include the following:

- PB: demineralized water to FPCC North only (Access to pools C & D initially)
- Recovery or Restoration of habitability conditions in the FHB to provide temporary access.

SFP cooling is available for maintaining the SFPs in an acceptable configuration unless (a) the accident sequence includes failures of support systems that affect SFP reliability, or, (b) the adverse conditions imposed by the event cause failure of the SFP cooling system or its supporting systems (e.g., AC power or CCW)

C.3 HEP DESIGNATORS

The HEP designator is structured similar to that used in the Level 1 analysis, but it also provides an indication of the severe accident conditions that, when imposed, cause variations in the HEP.

- Pre Initiator

  ¦ A ¦ O P ¦ C C C _ _ _ _

  A - System Involved in the Pre-initiator
  FPCC - F
  Demin. Water - D
  RWST - R
  Fire Protection - P
  NSW - N

  OP - Designator Pre-Initiator HEP
  CCC - Descriptive Portion of HEP

- Post Initiator

  ¦ O P E R ¦ A ¦ Q Q Q _ _ _ _

  OPER -Specifies a Post-Initiator HEP
  A - System Involved (see above)
  QQQ - Describes Action or Specifies the Number Designatior

C.4 PRE INITIATOR DESIGNATORS

Designator	Description
DOP-MISALIGN	Misalignment of Demin Water Precludes Success
EOP-MISALIGN	Misalignment of ESW Precludes Success
ROP-MISALIGN	Misalignment of RWST Precludes Success
FOP-MISALIGN	Misalignment of FPCC Precludes Success

Designator	Description
POP-MISALIGN	Misalignment of Fire Protection Precludes Success
NOP-MISALIGN	Misalignment of NSW Precludes Success
DOP-MISCAL	Common Cause Miscalibration of Demin. Sensors Causes Failure
EOP- MISCAL	Common Cause Miscalibration of ESW. Sensors Causes Failure
ROP- MISCAL	Common Cause Miscalibration of RWST Sensors Causes Failure
FOP- MISCAL	Common Cause Miscalibration of FPCC Sensors Causes Failure
POP- MISCAL	Common Cause Miscalibration of Fire Protection Sensors Causes Failure
NOP- MISCAL	Common Cause Miscalibration of NSW Sensors Causes Failure

C.5 POST INITIATOR DESIGNATORS

The following is a list of the critical operating crew actions that are included in the Spent Fuel Pool Assessment given that a Severe Accident has occurred which has led or will lead to containment failure or bypass.

Designator	Description	Procedure
OPER-TSC-E	TSC Fails to take pre-emptive action for early containment failures	None
OPER-TSC-L	TSC Fails to take pre-emptive action for late containment failures	None
OPER-IN-FA	Initiate FPCC Cooling to Pools A & B	OP116
OPER-IN-FC	Initiate FPCC Cooling to Pools C & D	OP116

Designator	Description	Procedure
OPERDALNPB	Align & Initiate Demin Water to FPCC for Makeup (PB)	OP116, 8.4
OPEREALNPA	Align & Initiate ESW to FPCC for Makeup (PA)	OP116, 8.13
OPERRALNPC	Align & Initiate RWST to FPCC for Makeup (PC)	OP116, 8.5
OPERMALNPD	Align & Initiate RMWST to FPCC for Makeup (PD)	OP116, 8.26
OPERDALNPF	Align & Initiate Demin Water FPCC Skimmer to FPCC for Makeup (PF)	OP116, 8.6
OPERRALNPG	Align & Initiate RWST to CLG pump to FPCC for Makeup (PG)	OP116, 8.12
OPERDALNPE(1)	Align & Initiate Demin Water FPCC Skimmer to FPCC for Makeup (PF)	OP116, 8.5
OPERPALNN1	Align & Initiate Fire Protection to FPCC for Makeup (N1)	None
OPERPALNN2	Align & Initiate Demin to Quick Disconnector to FPCC for Makeup (N2)	None
OPERPALNN3	Align & Initiate NSW to FPCC for Makeup (N3)	None
OPER-OFFST	Operators Fail To Use Portable/Off-Site Resources For Makeup To The SFPs	None
OPER-PROCD	Procedures To Maintain SFP Inventory Are Inadequate	ARP
OPER-GATE1	Operators Fail To Deflate Gate 1's Seals	None

(1) Not currently modeled.

Designator	Description	Procedure
OPER-GATE2	Operators Fail To Deflate Gate 2's Seals	None
OPER-GATE4	Operators Fail To Deflate Gate 4's Seals	None
OPER-GATE5	Operators Fail To Deflate Gate 5's Seals	None
OPER-GATE6	Operators Fail To Deflate Gate 6's Seals	None
OPER-GATE9	Operators Fail To Deflate Gate 9's Seals	None
OPER-GATES	Operators Fail To Remove Bulkhead Gates	None
OPER-1CLBA	Operators Fail To Cross Tie Unit 1 FPCCS Pump Train B To Heat Exchanger A	None
OPER-2CLBA	Operators Fail To Cross Tie Unit 2 FPCCS Pump Train B To Heat Exchanger A	None
OPER-ESW	Operators Fail To Open ESW Manual Valves into Fire Protection (e.g., seismic event)	None

C.6 PERFORMANCE SHAPING FACTORS

In the practice of HRA, it is generally agreed that qualitative analysis is the most important part, and that the benefits of quantification often may be relatively small. In terms of providing a sufficient basis for evaluation of system performance and possible suggestions for design changes, a qualitative analysis may in many cases be all that is needed.

The first step of an HRA is a task analysis or another type of systematic task description. Unless the task is known, it is impossible to appreciate the consequences of individual task steps and actions. The application of the method requires the identification of the scenarios or events for which a reliability analysis is needed. This will typically involve drawing up a comprehensive list of potential system failures that are serious enough to warrant further study. Such a list will include failures that reasonably can be expected, given the prior experience of the analyst with the type of system, the general operational experience, or the specific requirements imposed by the industry's regulatory body. This is normally done as part of the overall PSA, or as part of a more specific risk analysis. From this list, one particular scenario must be selected at a time as the focus for the analysis.

The performance shaping factors that dominate the assessment of operator response include the following:

- Time available and time required
- Stress
- Cues to initiate action
- Control Room Interface and availability of the Technical Support Center (TSC)
- Access to the areas (working conditions)
- Adequacy of training (e.g., JPMs)

- Competing tasks
- Complexity of tasks
- Procedures or guidance

Time Available and Time Required

The time required to perform most of the actions identified as capable of providing water makeup to the SFPs is estimated by Senior Reactor Operators (SROs) at 5 - 30 min. for manipulation. Additional times of 2 - 10 min for transit times are also estimated by SROs. Therefore, for HRA purposes it is considered prudent and consistent with the NRCs ASEP methodology [C-1] to double these estimates for time required. Therefore, the total required time is estimated on the order of 1 hour for most of the proposed options.

The time available for the crew, the TSC, the Operations Support Center (OSC), and offsite resources to take actions varies with the specific action and the accident sequence involved. Table C.6-1 summarizes some of the critical times available to take actions. The principal conclusion of the tabular analysis results are the following:

- ISLOCA sequences have an opportunity to provide effective mitigation to preserve the SFP water inventory conditions without heroic actions by access to FHB 216' El. North.
- This means all accidents have access to one or more pathways for alignment of makeup to the SFPs.
- If diesel fire pump (DFP) or demineralized water pumps are not available and portable pumps are required, then the windows for operator action vary such that late containment failures afford approximately 38 hours (but maybe as much as 90 hours) with essentially no on-site high radiation.
- Other accidents generally require working in a radiation environment, the severity of which depends on the accident type and the meteorology. The beneficial features of the site are that the local

areas where pathways can be aligned are spatially separated such that if the wind is carrying radiation to one location then other locations on-site would be affected to a lesser degree.

TABLE C.6-1

SUMMARY OF TIMES AVAILABLE FOR ACTIONS TO PRESERVE SFP WATER INVENTORY

Accident Sequence Type	Time of Adverse Radiation Condition		Time Available for Action		
	in RAB	in FHB	Align	Provide Pump Power: No radiation	Provide Pump Power: Radiation Work-around required
ISLOCA	0	0	96 hrs(2)/0 hrs	0	>16 hrs
SGTR	38-90 hrs	38-90 hrs	96 hrs(2)/38 hrs	0	Possibly from 38-96 hrs
Early Containment Failure	1 hr	1 hr	96 hrs(2)/0 hrs	1 hr	96 hrs
Late Containment Failure	38-90 hrs	38-90 hrs	96 hrs(2)/38 hrs	38-90 hrs	Possibly 38-96 hrs
Containment Isolation Failure	0	0(1)	96 hrs(2)/0 hrs	0	96 hrs

(1) Based on sensitivity case evaluation.

(2) Alignment in 216'El North (2SF201 manual valve) in the Demin System/Alignment in FHB 286' El.

Stress

The severe accident core melt progression would induce stress into the operating crew and other personnel in dealing with the severe accident. The characterization of this stress and its modeling is a difficult area that has been treated by Swain [4-2] in the evaluation of crew performance. Swain indicates that when overburdened by a situation, people react to stress in one or more of the ways listed below:

- Queueing – delaying some responses during overload, with the intention of responding at a later time.

- Omission – ignoring information or actions that are considered relatively unimportant.
- Gross discrimination – responding to gross aspects of signals and ignoring finer aspects, e.g., noting that the water level in the sump has risen but not noting the extent of the change.
- Errors – processing information incorrectly.
- Escape from task – physical or mental withdrawal.

Swain defines a stressor as "any external or internal force that causes bodily or mental tension." This definition allows an optimum level of stress as well as non-optimum levels. Reaction to a stressor is the stress that is felt. Stress *per se* is not undesirable. Unless there is some stress, nothing is likely to be accomplished in a work situation. Through common usage, the word "stress" has acquired a negative connotation because we tend to think of situations with high, incapacitating levels of stress. Dealing with stress, or even getting people to agree on what stress is, is not easy.

Figure C.6-1 from Swain [C-1] shows that when one plots stress level against performance effectiveness, the plot is not a linear one. With extremely high levels of stress (as exemplified by life-threatening emergencies), the performance of most people will deteriorate drastically, especially if the onset of the stressor is sudden and the stressing situation persists for long periods.

Figure C.6-1 also indicates that at very low levels of stress, performance will not be optimum. There is not enough arousal to keep a person sufficiently alert to do a good job. Under these conditions, some people tend to drowse on the job, or their level of attention and job involvement is materially reduced. The curve also shows that there is a level of stress at which performance is optimum. This optimum level of stress is difficult to define – it varies for different tasks and for different people, and is known as the Inverted Hypothesis or the Yerkes-Dobson Law.

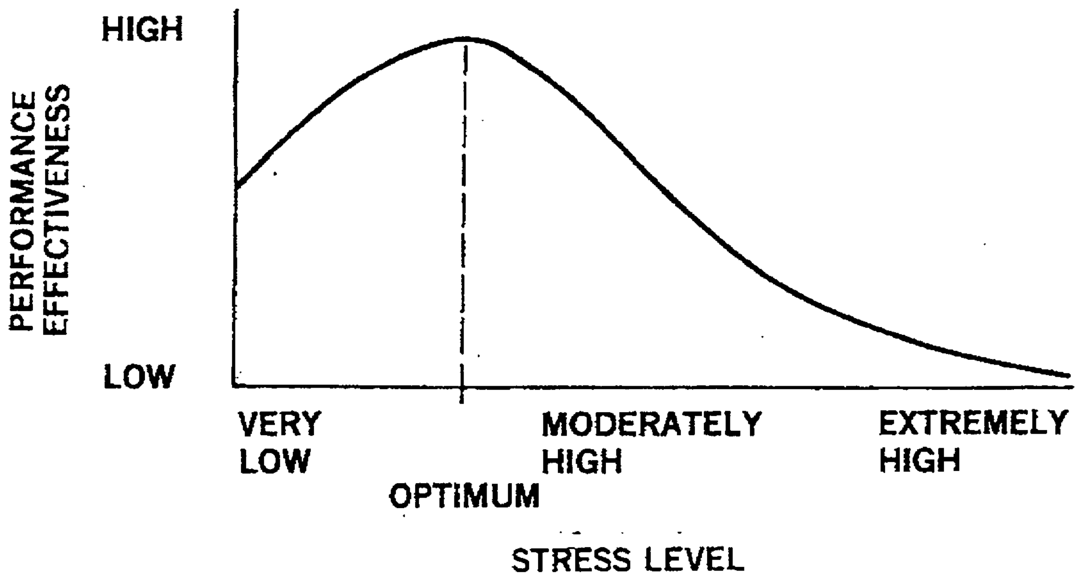

Figure C.6-1 Hypothetical relationship of psychological stress and performance effectiveness

The stress involved with response to a severe accident and to the need to protect the SFP is considered a high stress, but one that develops over an extended period of time where multiple personnel (operating crew, maintenance, plant management, and off-site resources) are all available to address the condition. As time progresses, the level of stress changes from the high, potentially debilitating stress to a more optimum stress level.

This extension in duration of the event to beyond a few hours is not explicitly treated by the Swain methodology. The tacit assumption is that with substantial time available to address a known (obvious), vitally important condition, the proper response will be taken when times longer than several hours are available.

Cues to Initiate Action and TSC Interface

The cues to initiate actions for SFP makeup have been included as two separate sensitivity cases:

A. TSC is required to be manned and have adequate guidance to direct prestaging

B. No action for alignment to be initiated unless the Annunciator Panel Procedures (APP) are entered.

Training

Training for the TSC and auxiliary operators is assumed to be average in the industry, i.e., well trained in the tasks required. No specific issues have been identified that would perturb the assessment as it affects the HEP calculations.

There is explicit training provided to the Auxiliary Operators (AOs)for the proceduralized local alignments related to the SFP (e.g., Watch Qualification Card requirements, or Job Performance Measures (JPMs)). Many of the alignments are part of normal operation, so they are performed by the AOs as part of their normal job function. Therefore, they have familiarity with the actions and procedures. The manipulations are considered "skill of the trade" and therefore no additional training is considered necessary. The one exception is the use of the demineralized water to the SFP through the suction of the FPCC when the pumps are not operating. This alignment is not strictly part of OP 116 and would need to be directed by the TSC given the current procedures at SHNPP.

Competing Tasks

Clearly there are competing tasks that will be on-going because the ASLB Order has specified a core melt accident progression with containment failure or bypass. This ongoing condition will result in expenditures of many resources to combat the causes of the problem and attempt whatever mitigation measures to limit the consequences.

Therefore, the SFP will likely be of a lower priority during the initial stages of a core melt progression event. Nevertheless, the amount of time available to prepare for the protection of the SFP is quite long and is considered by the two sensitivity cases A & B cited above.

As noted in Appendix A, the thermal hydraulic calculations indicate that the time for SFP boiling for the limiting case is ~ 20 hours. In addition, the time to boil away the SFP inventory and just begin to uncover spent fuel is estimated at 7 days (> 168 hours). These times can be compared with the severe accident times that have been calculated in the SHNPP PSA:

Accident Type	Time Relative to Accident Initiation	
	Containment Failure or Bypass	Initial Radionuclide Release
ISLOCA	0	0
SGTR	0	~ 1 Hr
Early Containment Failure	~ 2Hrs	~2 Hrs
Late Containment Failure	40-90 Hrs	40-90 Hrs

Procedures and Guidance

The proceduralized methods of SFP cooling and makeup to SFPs A & B are well written and clear. These procedures are considered to be characteristic of the procedures to be written for SFPs C & D when they become operational.

Access to Areas

If doses projected result in personnel exposure greater than 25 REM, the operator action is assumed not to be feasible for the purposes of analysis. This may be overly conservative. Failure probability is set to 1.0 for actions required under these conditions. (See discussion under interviews).

Local manual actions are not considered possible, if local temperatures greater than 120°F are encountered in the local area where no protection gear is worn.

However, CP&L has experience with protective clothing worn for fire fighting that clearly shows that surface temperatures on the protective suit in excess of 300°F can be tolerated by personnel. This factor primarily influences the cases with SFP boiling and where high temperatures may exist but without attendant high radiation. Therefore, using this protective gear, personnel would have access for periods of time sufficient to align fire hoses or connect demineralized water hoses to the SFP.

Other adverse conditions involving radiation release that may influence the operator action success are evaluated for each of the following accidents:

- I -- ISLOCA
- S -- SGTR
- E -- Early Containment Failure
- L -- Late or Very Late Containment Failure
- C -- Isolation Failure
- S -- Shutdown event with containment bypassed

The impact of these accidents is then input into the model through the use of Flag settings.

Because of the extended time available to respond to the boil off from the SFP, adverse conditions that may exist on site may have substantially subsided by the time the actions to protect the SFP arise.

Interview Input

In discussions with operating staff (former Shift Superintendent), the following items were identified as typical of Control Room response:

- No anticipatory actions (i.e., actions not in the Alarm Response Procedure) would be performed by the Control Room staff for the SFP. The restoration of SFP cooling is not considered an urgent action and will not be a priority action for the operating crew. For example, accidents with loss of SFP cooling would be responded to by restarting SFP cooling, but not establishing pool makeup. Other examples are that core damage, EOP or SAMG actions, or high temperature in the pool do not lead the Control Room crew to align makeup to the SFP. The TSC is expected to plan and arrange for actions involving the SFP loss of cooling.

- The accessibility to plant areas can be limited by radiation. If there are radiation levels in the specified areas that could lead to 5 REM exposure, the operator would not normally be released to perform the action. A cumulative dose of 25 REM could be authorized under extreme conditions. No action would likely be authorized for projected doses of greater than 25 REM.

C.7 QUANTIFICATION RESULTS

Table C-1 summarizes the quantified post-initiator operation actions that have been assessed for inclusion in the PSA to address the Postulated Sequence.

Table C-2 summarizes the HEPs modified to reflect the Case B Sensitivity Cases.

Table C-1

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS

New Basic Event	Base Case Prob.	Description	OP-116 Step
OPERDALNPB	1.90E-02	Operators Fail To Align DW To The Unit 1 FPCCS Cleanup Subsystem	8.4
OPERDALNPB	1.90E-02	Operators Fail To Align DW To The Unit 2 FPCCS Cleanup Subsystem	8.4
OPER-1CLBA	0.1	Operators Fail To Cross Tie Unit 1 FPCCS Pump Train B To Heat Exchanger A	N/A
OPER-2CLBA	0.1	Operators Fail To Cross Tie Unit 2 FPCCS Pump Train B To Heat Exchanger A	N/A
OPERPALNN1	6.20E-02	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	N/A
OPER-GATE1	1	Operators Fail To Deflate Gate 1 Seals	N/A
OPER-GATE2	1	Operators Fail To Deflate Gate 2 Seals	N/A
OPER-GATE3	1	Operators Fail To Deflate Gate 3 Seals	N/A
OPER-GATE4	1	Operators Fail To Deflate Gate 4 Seals	N/A
OPER-GATE5	1	Operators Fail To Deflate Gate 5 Seals	N/A
OPER-GATE6	1	Operators Fail To Deflate Gate 6 Seals	N/A
OPER-GATE7	1	Operators Fail To Deflate Gate 7 Seals	N/A
OPER-GATE9	1	Operators Fail To Deflate Gate 9 Seals	N/A
OPER-GATES	1	Operators Fail To Remove Bulkhead Gates	8.27
OPERPALNN2	1	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	N/A
OPERPALNN3	1	Operators Fail To Use Water From The NSW System In The WPB To Makeup To The SFP	N/A
OPER-OFFST	0.1	Operators Fail To Use Portable / Off-Site Resources For Makeup To The SFPs	N/A

Table C-1

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS

New Basic Event	Base Case Prob.	Description	OP-116 Step
OPER-PROCD	1.00E-03	Procedures To Maintain SFP Inventory Are Inadequate	All
OPERRALNPC	1	Operators Fail To Align The FPCCS Purification Subsystem To The RWST	8.5
OPER-LOLVL	1.00E-03	Operators Fail To Diagnose Low SFP Levels And / Or Perform Recovery	All
OPER-ESW	0.1	Operators Fail To Open ESW Manual Valves	8.13
OPER-TSC-E	4.60E-03	TSC Fails to Take Pre-emptive Action for Early Failures	NA
OPER-TSC-L	2.40E-03	TSC Fails to Take Pre-emptive Action for Late Failures	NA
OPER-SKIMR	1	Operators Fail To Open The Crosstie Between Units 1&4 and 2&3 FPCCS Skimmers	NA
OPER-DWXTM	1	Operators Fail To Open DM Crosstie Valve 1SF-203	NA
OPER-START	2.00E-05	OPERATORS FAIL TO MANUALLY START FPCS MOTOR-DRIVEN PUMP	NA
OPERZOFFST	5.00E-02	Operator Fails to Align Offsite Resources to Previously Established Paths	NA
CI-CASE 1 (Shutdown)	1.1 E-2	Operator Fails to Restore Primary Containment Given Mid Loop Operation (Shutdown only)	Tech Specs
CI-CASE 2 (Shutdown)	1.6 E-2	Operator Fails to Restore Primary Containment Given Normal Level	Tech Specs
<u>Operator Actions Not Credited in this Analysis</u>			
OPEREALNPA	1	Operator Fails to Align and Initiate ESW to FPCC for Makeup	8.13
OPERMALNPD	1	Operator Fails to Align and Initiate RMWST to FPCC for Makeup	8.26
OPERDALNPE	1	Operator Fails to Align and Initiate Demin Water to FPCC Skimmer for Makeup	8.6

Table C-1

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS

New Basic Event	Base Case Prob.	Description	OP-116 Step
OPERRALNPF	1	Operator Fails to Align and Initiate RWST to FPCCS Cooling Pump for Makeup	8.5
OPERDALNPG	1	Operator Fails to Align and Initiate Demin Water to FPCC Cleanup for Makeup	8.5
OPER-IN-FA	1	Operator Fails to Initiate FPCC Cooling to Pools A and B	N/A
OPER-IN-FC	1	Operator Fails to Initiate FPCC Cooling to Pools C and D	N/A

Table C-2

SHNPP SFPAET HEP'S AS SENSITIVITY INPUTS

Basic Event	Description	Base Case	Case B
OPERDALNPB	Operators Fail To Align DW To The Unit 1 or Unit 2 FPCCS Cleanup Subsystem	1.90E-02	9.50E-03
OPER-TSC-E	TSC Fails to Take Pre-emptive Action for Early Failures	1.30E-02	2.30E-03
OPERPALNN1	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	6.20E-02	1.10E-03
OPERPALNN2	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	1.00E-00	2.50E-01
OPER-TSC-L	TSC Fails to Take Preemptive Action for Late Failure	2.40E-03	1.40E-03

REFERENCES

[C-1] Swain, A.D., Guttmann, H.E., Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, August 1983.

[C-2] Parry, G.W., An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment, EPRI TR-100259, June 1992.

[C-3] Swain, A.D., Accident Sequence Evaluation Program Human Reliability Analysis Procedure, NUREG/CR-4772, SAND86-1996, February 1987.

Appendix D

SPENT FUEL POOL ASSESSMENT EVENT TREE (SFP-AET)

The purpose of this appendix is to describe the accident sequence event tree model for the evaluation of adequate cooling for the fuel located in the Spent Fuel Pools (SFPs), given a severe accident that fails or bypasses containment. The Spent Fuel Pool Assessment Event Trees (SFP-AET) are used to characterize the accident progression effects that may compromise the ability to maintain coolable conditions in the SFPs. Figure D-1 is the SFP-AET. Subsections D.1 through D.10 discuss the structure of the SFP-AET and each of the top events in the event tree.

The success criteria used in the SFP assessment are discussed first.

<u>Success Criteria</u>

The probabilistic model has been structured in a realistic manner. In addition, the success criteria for the model are also based on a realistic assessment with the following exceptions:

- SFPs C & D are the focus of the evaluation. However, pools A & B may lose inventory prior to pools C & D given certain severe accidents. The consequences of loss of inventory to pools A & B may in turn adversely impact both access and further preventative actions related to pools C & D. Therefore, the success criteria have been structured to require adequate makeup or cooling of all 4 pools. From the standpoint of the ASLB Order, this assumption regarding success criteria may introduce some potential conservatisms.

- The limiting heat load to the SFP is generally that in pools A & B. This is where the fuel with the highest decay heat levels is present. Refer to Table D-1 and the discussion below.

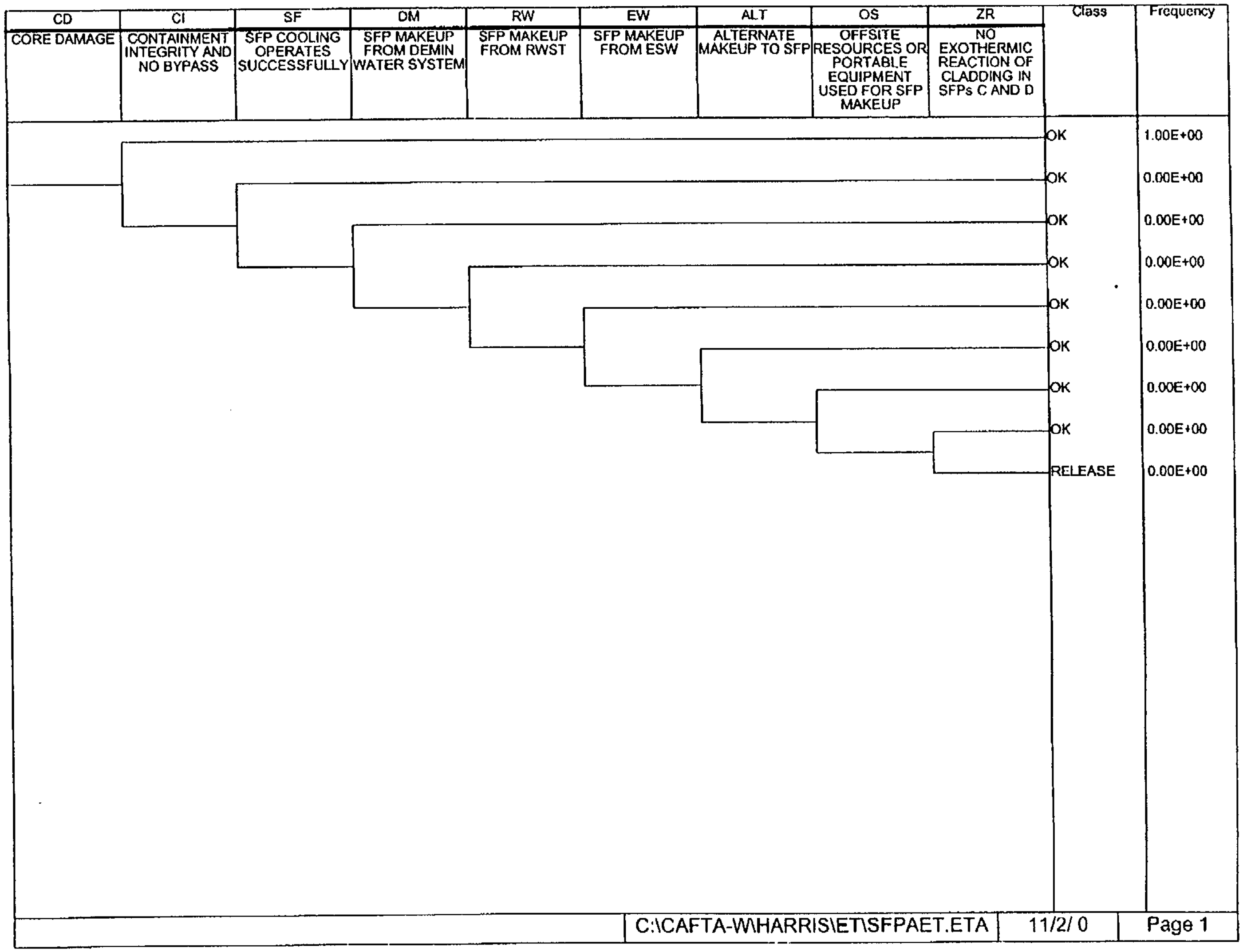

Figure D-1 Spent Fuel Assessment Tree (SFP-AET)

- The limiting heat load is predicated on the plant practice of discharging 1/3 of the core into spent fuel pool A as part of a core shuffle. An integration of the times to boil and uncover spent fuel as a function of the time following a full core offload could be performed to assess the available time for effective operator action. This integration or weighted averaging has not been performed to avoid obscuring the results of the analysis. The assumption of the peak heat load in the pools results in a conservative assessment of times available to achieve successful alignment of water makeup sources. This situation, however, exists for only short periods of time. Nevertheless, the analysis considered the limiting heat load in pool A as <u>always</u> present.

- Makeup to the SFPs is assessed to be aligned to only one pool. This requires sufficient makeup volume and flow rate to overflow the pool gates and spill into the transfer canals and the other pools to maintain adequate inventory in <u>all</u> pools.

  This is a conservative assumption, but is believed not to significantly bias the resulting assessment, i.e., the analysis is believed realistic.

The results of CP&L calculations are summarized in Table D-1.

Table D-1

SFP Conditions for Various Assumed Decay Heat Levels

Pools	Time to reach boiling temperature	Additional time for water level to reach top of racks	Total time	Makeup required to offset boiling
A and B (Beginning of cycle)	20.57 hours	7.21 days	8.07 days	53.70 gpm
A and B (End of cycle)	38.67 hours	13.56 days	15.17 days	28.57 gpm
C and D (1 MBTU/hr heat load)	384.66 hours	99.99 days	116.02 days	2.15 gpm
C and D (15.6 MBTU/hr heat load)	34.42 hours	8.80 days	10.23 days	33.64 gpm

These calculations did not take credit for any additional cooling or makeup that would be available.

For cases with the SFPs near capacity, SFPs C and D would have a slightly larger heat load than SFPs A and B when SFPs A and B are examined at the end of a fuel cycle.

The following is a discussion of the event tree nodes for the SFP-AET, Figure D-1.

D.1 CD: CORE DAMAGE

The first node is the input to the SFP analysis, i.e., the frequency of the first two steps specified by the ASLB Order. For internal events, this includes the transfer of the cutsets from the Level 1 and 2 PSA which describe those failure events that could lead to a core damage event plus containment failure or bypass.

D.2 CI: CONTAINMENT INTEGRITY AND NO BYPASS

The second node is provided solely to show that cutsets of interest are those associated with both core damage and the containment failed or bypassed. If the containment is intact (success branch), the sequences are not analyzed because those accident sequences are not part of the ASLB Order.

Success Criteria

The success criteria for this branch is that the containment has been successfully isolated and no containment failure or bypass has occurred.

The criteria for failure of containment is established as part of the Level 1&2 Harris PSA. No changes have been made to those criteria.

Up Branch (Success)

Containment intact and not bypassed leads to sequences that are not part of the ASLB Order and therefore no additional analysis is performed.

Down Branch (Failure)

The down branch of the event tree node leads to conditions which have radionuclide releases outside containment. The effect of these releases on systems and accessibility are evaluated in the subsequent event tree nodes.

Figure D.2-1 is the Functional Fault Tree for the Containment Integrity mode. This node is forced by the quantification to be reflective of the accident type that is input into the evaluation (e.g., ISLOCA, SGTR, etc.).

The evaluation includes the following accident types, all of which involve containment failure or bypass. Their specific timing and release paths are treated explicitly in the evaluation of the subsequent event tree nodes:

- ISLOCA
- Containment failure early
- Containment failure late
- Steam Generator Tube Rupture (SGTR)
- Containment Isolation Failure

D.3 SF: SPENT FUEL POOL COOLING OPERATES SUCCESSFULLY

The Fuel Pool Cooling and Cleanup System (FPCCS) cooling subsystem is the primary method of maintaining the SFPs in a safe condition. In addition, the large water inventory of the pools provides substantial time available to restore FPCCS cooling if it should be interrupted.

The SFPs are maintained in a condition to cool the spent fuel by virtue of the FPCCS cooling subsystem. There are two separate FPCCS cooling subsystems. The two FPCCS cooling subsystems are arranged to provide cooling to:

(a) Pools A and B via the "Units 1 and 4" FPCCS cooling subsystem; and,

(b) Pools C and D via the "Units 2 and 3" FPCC cooling subsystem. (New system to be installed.)

Each of the FPCCS cooling subsystems are composed of the following:

- 2 FPCCS cooling pumps;
- 2 FPCCS cooling heat exchangers;
- Component Cooling Water (CCW) cooling of the heat exchangers; and,
- AC Power support from 1A-SA and 1B-SB for the A and B pump trains, respectively.

This model logic addresses the following types of failure modes:

- FPCCS cooling failures (random, human error, test/maintenance and common cause);
- FPCCS cooling support system failures including support system failures that may have contributed to the core damage accident sequence cutsets identified in the first node of the event tree; and,
- Consequential failures of FPCCS cooling or its support systems due to adverse environmental conditions caused by containment failure or bypass.

Success Criteria

The success criteria for this node is that sufficient SFP cooling is available to prevent boiling in all four pools.

Up Branch (Success)

Establishing cooling to all four pools is considered a successful end state. No additional analysis of the SFPs is required.

Because the pools are normally isolated by installed bulkhead gates, A & B pools require cooling AND C & D pools require cooling.

Down Branch (Failure)

The failure of the FPCCS cooling subsystems to provide adequate makeup to the SFPs requires the evaluation of additional methods of makeup.

Figure D.3-1 is the top logic fault tree describing the failure modes considered for the FPCCS cooling subsystem.

D.4 DM: SPENT FUEL POOL MAKEUP FROM DEMINERALIZED WATER SYSTEM

Even if the FPCCS cooling subsystem fails and is not capable of cooling the SFPs, there are many methods of providing inventory makeup to the SFPs. The makeup methods are varied and diverse. They provide the capability to establish adequate water inventory to keep the spent fuel covered and therefore cooled. The various methods are discussed in Subsections D.4 to D.8.

The first of the methods involves the use of the demineralized water system.

Figure D.4-1 is the top logic functional fault tree for demineralized water makeup to the SFPs.

The fault tree considers the following in its identification of failure modes:

- Hardware failures including random, test / maintenance, human error and common case failures
- Hardware failures caused by adverse environmental conditions resulting from containment failure or bypass
- Support system failures
  - Loss of off-site AC Power
  - Loss of makeup water source
- Adverse conditions resulting from containment failure or bypass that may preclude local alignment actions by the operating crew
  - Adverse environmental conditions
  - Radiation fields that could prevent local access to the areas for required alignment
- Failure of recovery actions such as AC Power restoration

<u>Success Criteria</u>

The success criteria for this node is that sufficient demineralized water injection is available to perform the following:

- Makeup greater than 100 gpm;
- Makeup to all of the four pools; and,
- Volume available of more than 66,000 gal. (Demin Water Tank has substantially more volume available.)

Down Branch (Failure)

If the demineralized water injection path or pumps are unavailable to meet the success criteria, this node is failed. The failure of the demineralized water system to provide adequate makeup to the SFPs requires the evaluation of additional methods of makeup.

D.5 RW: SFP MAKEUP FROM THE RWST

SFP makeup from the Refueling Water Storage Tank (RWST) is a useable method of makeup under most plant conditions. The specific low frequency severe accident sequences being considered here, however, may result in transferring the contents of the RWST into containment. Therefore, the viability of this protection method for the SFPs is accounted for in the quantification of the model by setting its unavailability to 1.0.

Figure D.5-1 is the top logic functional fault tree describing the failure modes of makeup to the SFPs from the RWST.

The fault tree considers the following in the identification of failure modes:

- Hardware failures including random, test / maintenance, human error and common case failures
- Hardware failures caused by adverse environmental conditions resulting from containment failure or bypass
- Support system failures
    - Loss of makeup water source
- Adverse conditions resulting from containment failure or bypass that may preclude local alignment actions by the operating crew
    - Adverse environmental conditions

- Radiation fields that could prevent local access to the areas for required alignment

- Failure of recovery actions such as AC Power restoration

Success Criteria

The success criteria for this node is that sufficient RWST injection is available to perform the following:

- Makeup greater than 100 gpm;
- Makeup to all four pools; and,
- Volume available of more than 66,000 gal.

Up Branch (Success)

Establishing makeup from the RWST is considered a successful end state. No additional analysis of the SFPs is required.

Down Branch (Failure)

The failure of RWST to provide adequate makeup to the SFPs requires the evaluation of additional methods of makeup. The specific low frequency severe accident sequences being considered here however may result in transferring the contents of the RWST into containment. Therefore, the viability of this protection method for the SFPs is accounted for in the quantification of the model by setting its unavailability to 1.0.

D.6 FW: SFP MAKEUP FROM ESW

The Emergency Service Water (ESW) System is a potential large volume makeup method to the SFPs. ESW supplies water from Shearon Harris Lake. ESW is not normally connected to FPCCS. Operators must enter the Reactor Auxiliary Building

(RAB) on the 236' elevation and connect a 50' rubber hose to between two manual valves to establish this connection.

Figure D.6-1 is the top logic functional fault tree for the makeup from ESW to the SFPs.

The fault tree considers the following in the identification of failure modes:

- Hardware failures including random, test/maintenance, human error and common case failures
- Hardware failures caused by adverse environmental conditions resulting from containment failure or bypass
- Support system failures
  - Loss of off-site AC Power
  - Loss of makeup water source
- Adverse conditions resulting from containment failure or bypass that may preclude local alignment actions by the operating crew
  - Adverse environmental conditions
  - Radiation fields that could prevent local access to the areas for required alignment
- Failure of recovery actions such as AC power restoration

<u>Success Criteria</u>

The success criteria for this node is that sufficient ESW injection is available to perform the following:

- Makeup greater than 100 gpm;
- Makeup to all four pools; and,
- Volume available of more than 66,000 gal. (An almost inexhaustible supply is available from SHNPP Lake.)

Up Branch (Success)

Establishing makeup from ESW is considered a successful end state. No additional analysis of the SFPs is required.

Down Branch (Failure)

The failure of the ESW to provide adequate makeup to the SFPs requires the evaluation of additional methods of makeup.

D.7 ALT: ALTERNATE MAKEUP TO SFP

Alternate makeup sources to the SFPs given that the proceduralized alignments are ineffective consist of the following:

- Fire Protection System via hose stations on the 286' elevation of the FHB;
- Demineralized Water System via quick connect hoses on Elevation 286' of the FHB; or,
- Normal Service Water to the pools via a rubber hose from a header in the Waste Processing Building.

All of these alternate methods of makeup to the SFP are treated in the model evaluation.

Figure D.7-1 is the top logic functional fault tree for the alternate injection methods.

The fault tree considers the following in the identification of failure nodes:

- Hardware failures including random, test/maintenance, human error and common case failures

- Hardware failures caused by adverse environmental conditions resulting from containment failure or bypass

- Support system failures

  - Loss of off-site AC power
  - Loss of makeup water source

- Adverse conditions resulting from containment failure or bypass that may preclude local alignment actions by the operating crew

  - Adverse environmental conditions
  - Radiation fields that could prevent local access to the areas for required alignment

- Failure of recovery actions such as AC power restoration

Success Criteria

The success criteria for this node is that sufficient alternate injection is available to perform the following:

- Makeup greater than 100 gpm;

- Makeup to all four pools; and,

- Volume available of more than 66,000 gal. (All sources considered have substantially more volume available for injection.)

Up Branch (Success)

Establishing makeup from the Alternate System is considered a successful end state. No additional analysis of the SFPs is required.

Down Branch (Failure)

The failure of the alternate sources to provide adequate makeup to the SFPs requires the evaluation of additional methods of makeup.

D.8 OS: OFFSITE RESOURCE OR PORTABLE EQUIPMENT USED FOR SFP MAKEUP

The time for overheating of the fuel in the SFPs due to evaporation is quite long; the minimum time has been estimated to be seven days. This means that off-site resources have substantial time to be organized and provided on-site. Of course, any on-site radiation that could affect access will need to be addressed. However, given the long times available, it is considered likely that methods of restoring access for short periods of time can be formulated by the TSC team.

The primary methods considered as part of the offsite resources are:

- Portable pumps and small electric generators that can be trucked in or airlifted to the site to provide suction from the intake or cooling water basin into pre-aligned pathways (demineralized water or fire protection).

- Fire pumper truck to perform similar activities.

It is noted that during 1999 for an approximate 2 week period, the Holly Springs Fire Department provided two pumper trucks and 24 hours a day coverage to meet the procedural requirements specified in the SHNPP procedure—FPP-013 Fire Protection-Minimum Requirements and Mitigating Actions --during an equipment outage. In addition, the Holly Springs Fire Department participated in the last drill under the emergency plan.

The Apex Fire Department is the closest fire department (approximately five miles from the plant, across US-1). This department is under contract to CP&L. Holly Springs and Fuquay-Varina Fire Departments are also under contract with CP&L. Holly Springs and Fuquay fire support would access the plant from opposite directions and would not cross US-1.

Success Criteria

The success criteria for this node is that sufficient injection is available to perform the following:

- Makeup greater than 100 gpm;
- Makeup to all four pools; and,
- Volume available of more than 66,000 gal.

Up Branch (Success)

Establishing makeup to the SFPs using resources from off-site is considered a successful end state. No additional analysis of the SFPs is required.

Down Branch (Failure)

In the event that no on-site resources are available, the failure of off-site sources to provide adequate makeup to the SFPs could result in uncovery of the spent fuel.

D.9 ZR: NO EXOTHERMIC REACTION OF CLADDING IN SFPs C&D

The C&D Fuel Pools will contain fuel that has been removed from the reactor for more than five years. This means the decay heat levels are quite low. As a result of this low decay heat level and despite the evaporation of water surrounding the spent fuel in Pools C and D, there is a high probability that the fuel will remain adequately cooled by heat transfer to the air.

This event tree node is used in this analysis to demonstrate the sensitivity of the calculated results to the assertion that a Zircaloy (ZR) exothermic reaction could occur releasing fission products from Pools C and D.

Figure D.9-1 is the top logic functional fault tree describing the failure possibilities.

Success Criteria

The success criteria for their branch is that the spent fuel in Pools C and D can be air-cooled and avoid exothermic ZR reactions.

There are a number of important aspects of the ZR exothermic interaction. These include the following:

- Air cooling of the fuel in the C&D Fuel Pools has been assessed by Sandia and Brookhaven National Laboratories (SNL and BNL) to be feasible when the fuel has been removed from the reactor for more than five years.
- Speculation regarding other fuel and clad conditions (e.g., hydriding) that could result in more adverse conditions than identified by SNL or BNL leads to postulated clad exothermic reactions for a spent fuel uncovery.

Up Branch (Success)

Successful air cooling prevents a radionuclide release from Pools C and D.

Down Branch (Failure)

The down branch represents failure to adequately cool the spent fuel in Pools C and D. This represents Step 7 of the Postulated Sequence specified in the ASLB.

For the purposes for the base case assessment, a conditional failure probability of 1.0 was assigned to this step of the Postulated Sequence. This node is also used as part of a sensitivity evaluation to demonstrate the variation in the overall Postulated Sequence frequency.

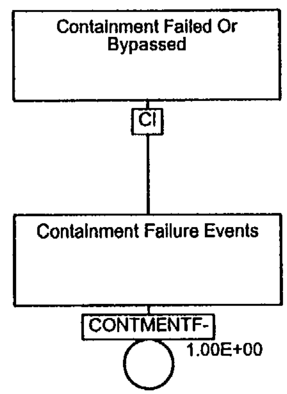

Figure D.2-1 Functional Fault Tree for Containment Isolation

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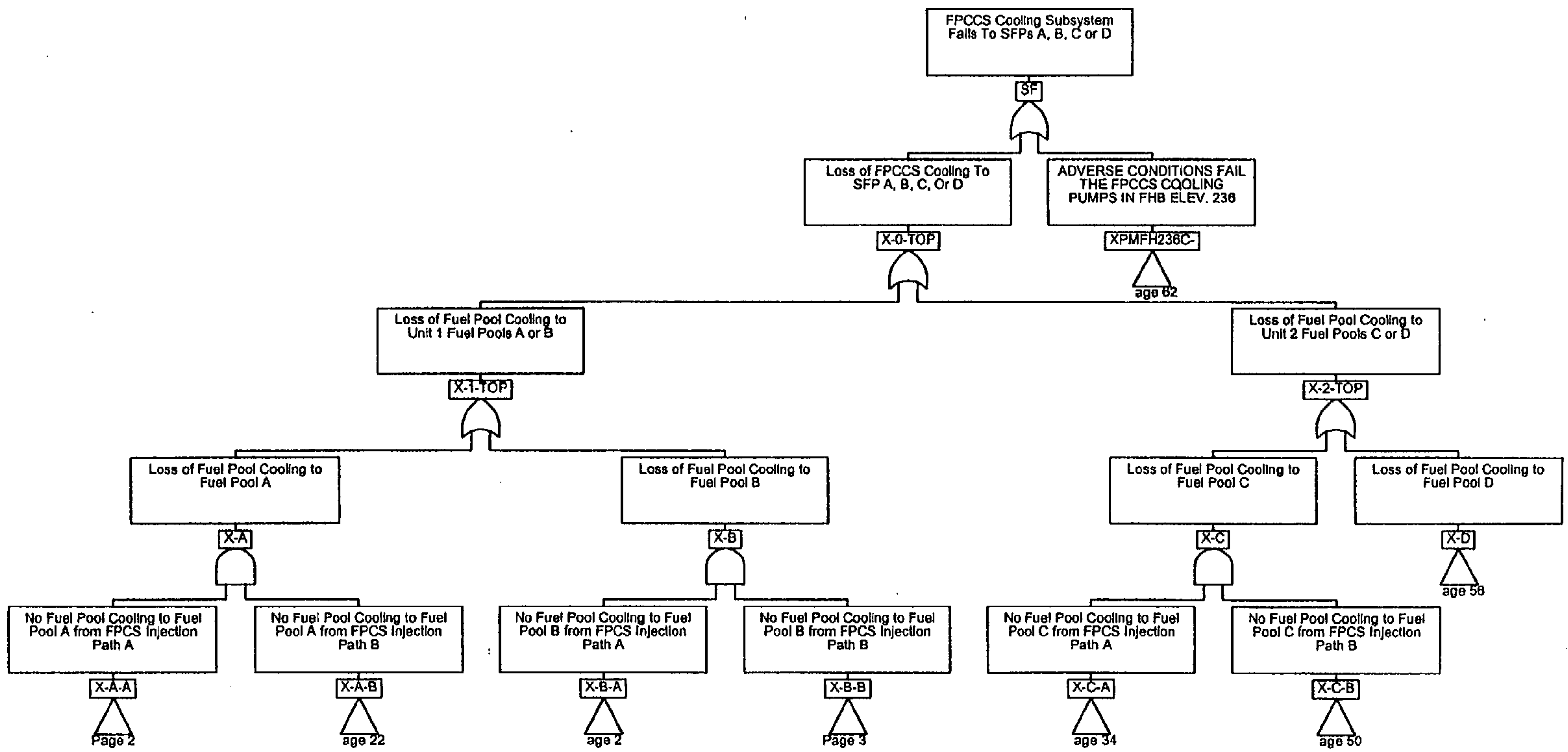

Figure D.3-1 Functional Fault Tree for the FPCCS Cooling System

1	2	3	4	5	6

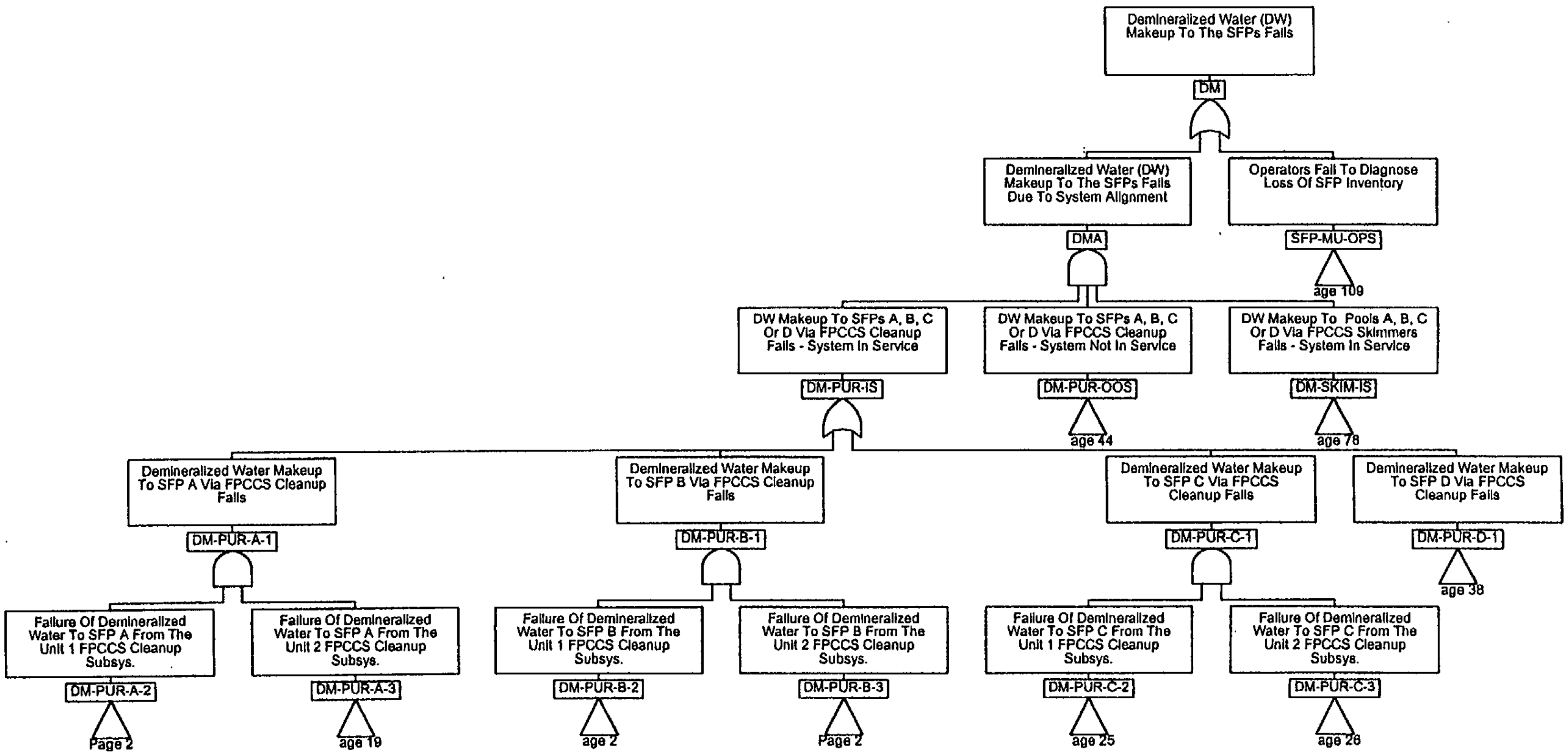

Figure D.4-1 Functional Fault Tree for Demineralized Water Makeup to SFP

1	2	3	4	5	6

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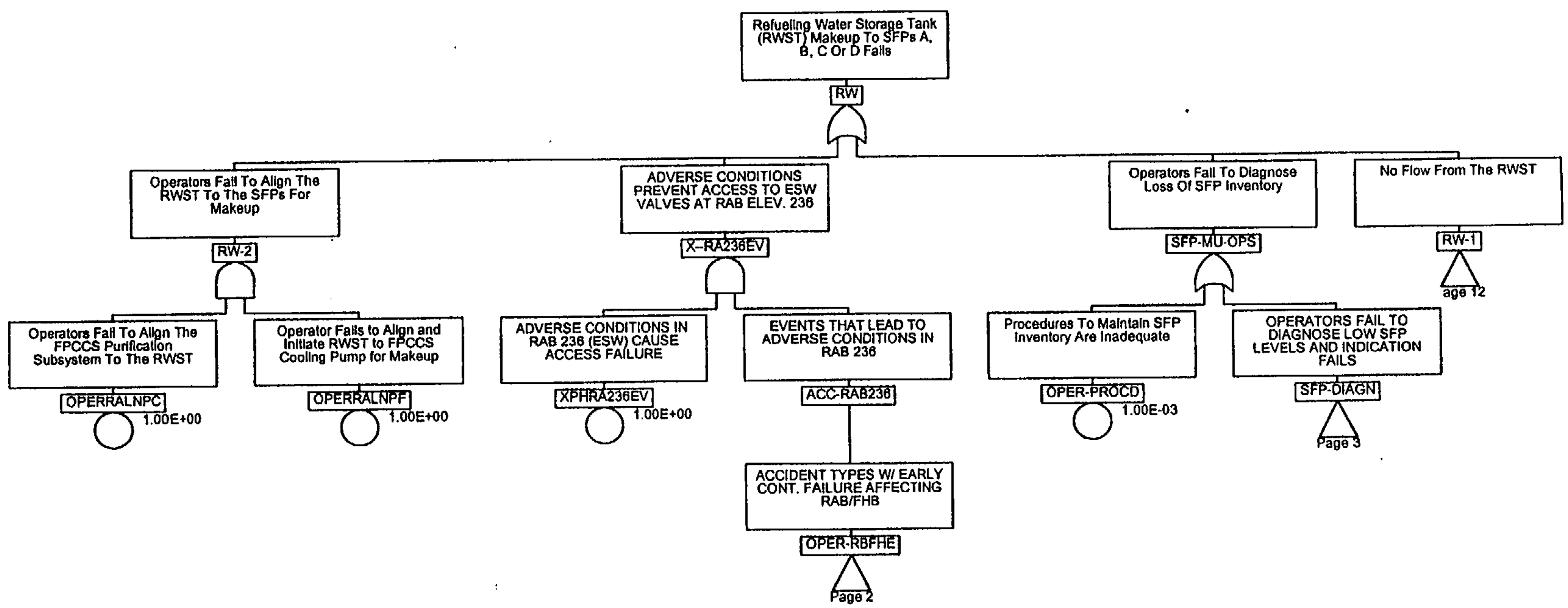

Figure D.5-1 Functional Fault Tree for RWST Makeup to SFP

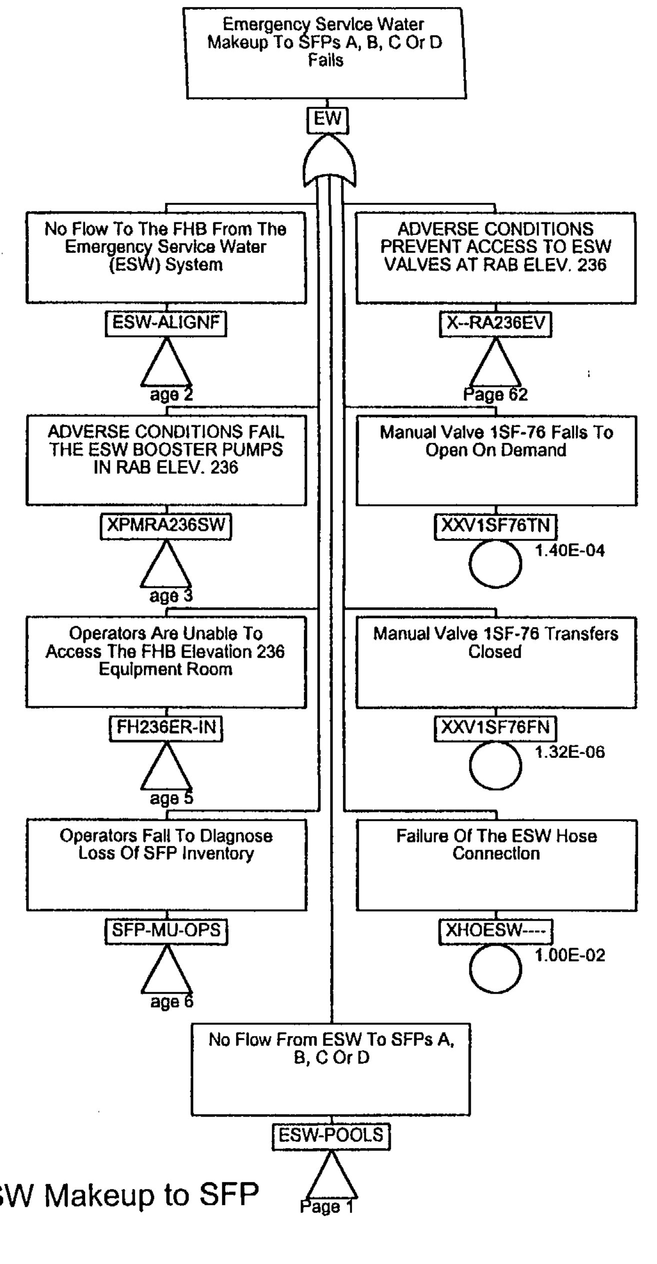

Figure D.6-1 Functional Fault Tree for ESW Makeup to SFP

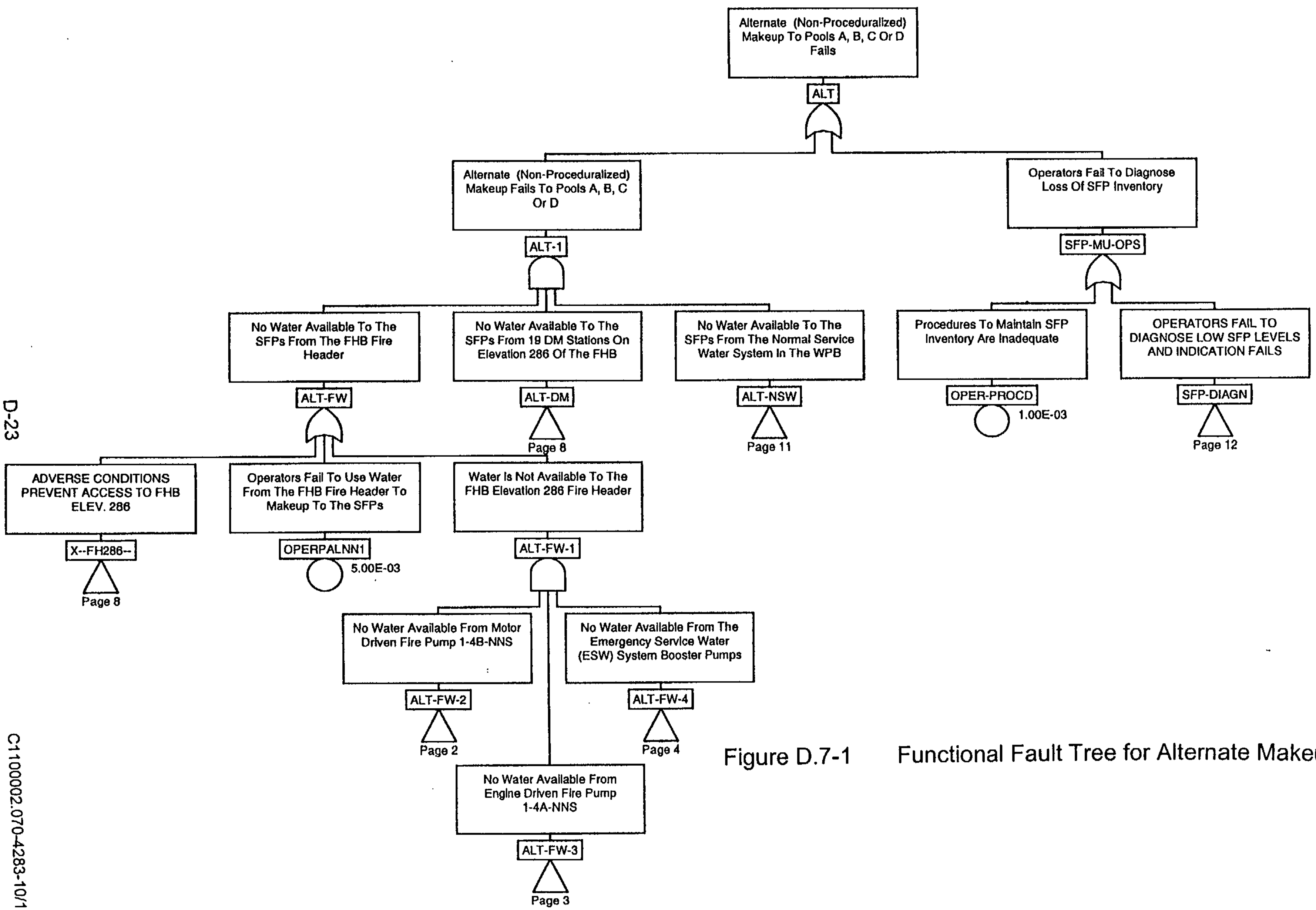

Figure D.7-1 Functional Fault Tree for Alternate Makeup to SFP

D-23

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1	2	3	4	5	6

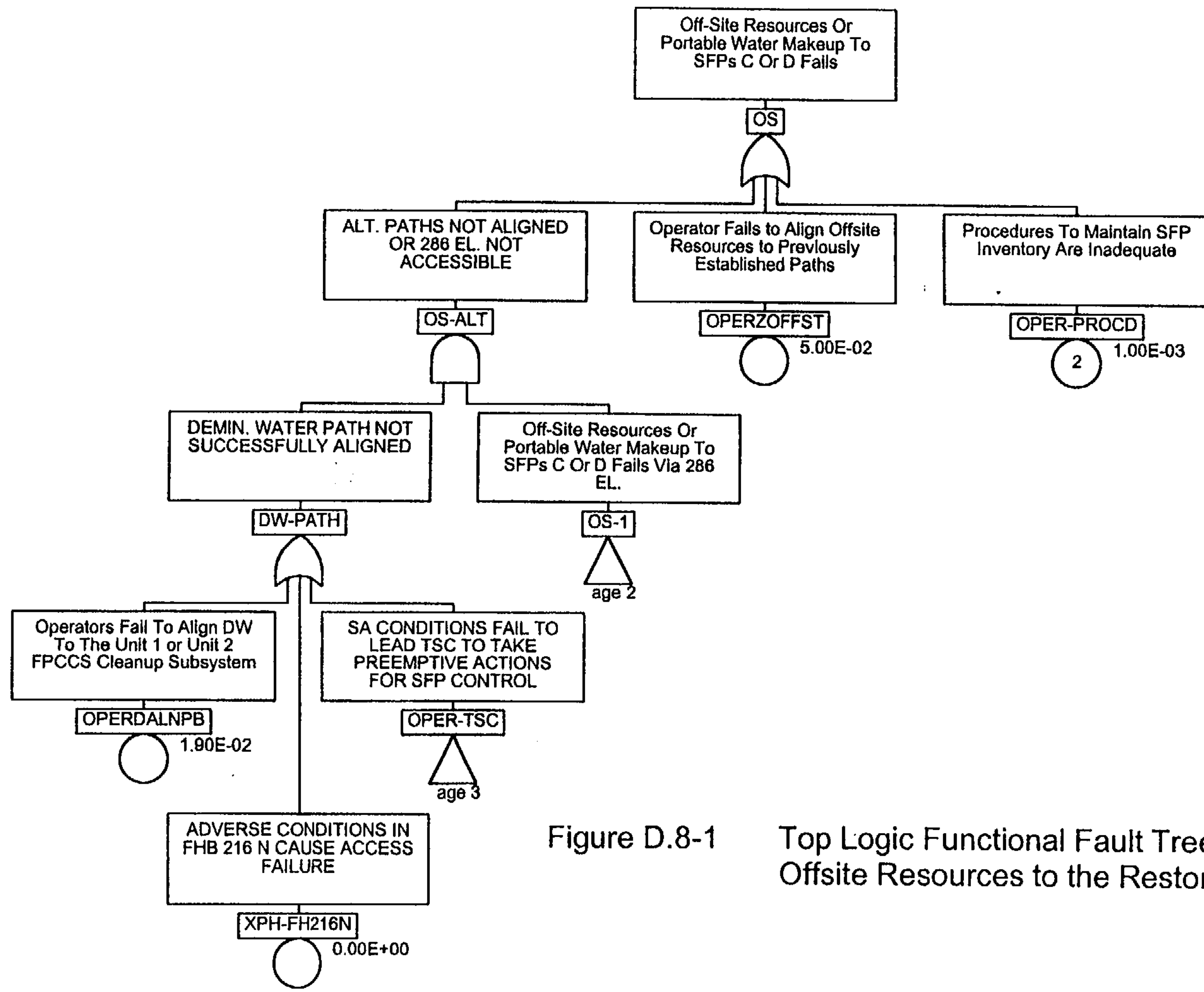

Figure D.8-1 Top Logic Functional Fault Tree for Application of Offsite Resources to the Restoration of SFP Makeup

1	2	3	4	
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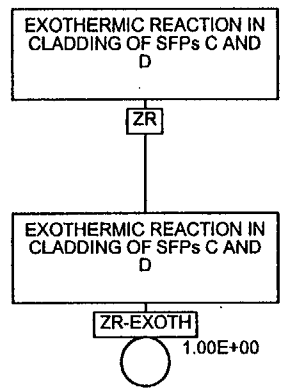


Figure D.9-1 Functional Fault Tree Defining the Logic for Zircalloy Exothermic Reaction in Air That Leads to Release of Radionuclide

1

SHNPP SFP Assessment	T:\HARRIS\TADSFP~1\SFP.CAF	10/11/ 0	Page 1

Appendix E

DETERMINISTIC ANALYSIS

MAAP ANALYSES

E.1 BACKGROUND

Selected analyses have been carried out using the Modular Accident Analysis Program (MAAP) [E-1]. These calculations are aimed at providing the thermal-hydraulic response of the Reactor Auxiliary Building (RAB) and the Fuel Handling Building (FHB). The RAB and FHB conditions following a postulated severe accident will impact the ability of operators to enter these buildings to take certain mitigative actions, and for the equipment to survive the environmental conditions.

In order to perform the MAAP analyses, a Harris specific parameter file developed in 1992 was used. This is the same set of parameters utilized in the original Individual Plant Examination (IPE) [E-3] analyses. The parameter file includes approximately 1000 plant-specific inputs that define the primary system, containment, and ECCS systems at the plant. In order to extend the analysis to include a representation of the RAB and FHB, the Harris parameter file was expanded utilizing the node and junction auxiliary building model in MAAP. Section A.2 provides a description of the RAB and FHB modeling.

Details of MAAP, including the auxiliary building model, can be found in the MAAP Users Manual [E-2].

A series of 5 scenarios were analyzed with MAAP to investigate the RAB and FHB response to postulated severe accident conditions. Included in these core damage scenarios are:

1. Interfacing System LOCA
2. Steam Generator Tube Rupture
3. Containment Isolation Failure
4. Vessel breach with Early Containment Failure
5. Vessel breach with Late Containment Failure

An additional calculation was performed to investigate the temperature response of the FHB under spent fuel pool (SFP) boiling conditions.

Section E.3 will provide the detailed results for the selected MAAP analyses.

E.2 REACTOR AUXILIARY BUILDING AND FUEL HANDLING BUILDING MODEL DEVELOPMENT

The following describes the development of a MAAP 3.0B model to represent the SHNPP RAB and FHB. The focus of the model is to predict the thermal-hydraulic and radionuclide environment in key RAB and FHB compartments to support the SFP evaluation. The current MAAP 3.0B auxiliary building model limits the total number of control volumes to 9. To address the conditions in the key areas of the RAB and FHB, the following nodalization was developed.

The two plant walkdowns reported in Appendix F provided substantial insights into the RAB and FHB building layout and the expected response to severe accident conditions.

Node #	Building	Elevation	Description
1	RAB	190'	NE quadrant representing possible ISLOCA location
2	RAB	216'	East section
3	RAB	235'	East section adjacent to containment
4	RAB	236'	Remaining section of RAB including CCW pumps and heat exchangers
5	RAB	261' + 286'	Upper elevations of RAB
6	FHB	216'	North
7	FHB	216'	South
8	FHB	236'	Center section including fuel pool cooling pumps and heat exchanges
9	FHB	261' + 286'	Operating deck

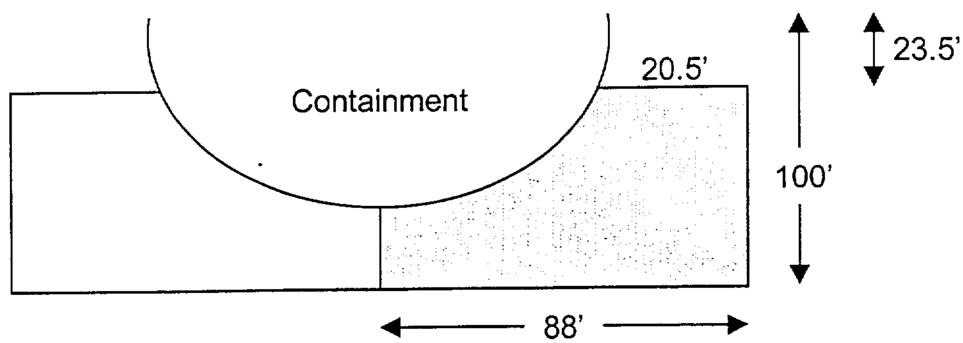
The following describes each control volume. Wall heat sinks have been conservatively estimated using only the outer perimeter of each node. No credit is taken for internal walls and equipment.

Node 1

RAB El. 190'

Potential break location for ISLOCA

Reference: CAR-2165 G-015



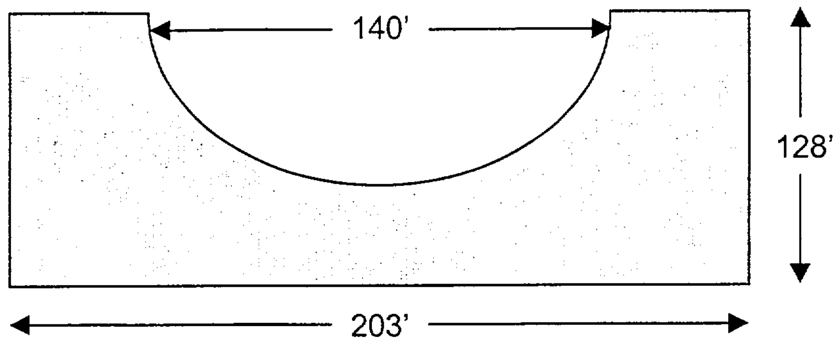
From G-013, ID of Containment	= 130'
At El. 190' assume wall thickness	= 5'
Therefore, OD of containment	= 140'
¼ of Containment is	= 3,848 ft^2
Excluded section is	= 23.5 x 20.5 = 482 ft^2
Floor Area	= 100 x 88 – 3,848 – 482 = 4,470 ft^2
Height	= 216 – 190 = 26'
Volume (Assume free volume is 90% of total volume to account for structures)	= 4,470 x 26 x .90 = 104,598 ft^3
Heat Sink Area	= (88 + 76.5) x 26 = 4,277 ft^2

Node 2

RAB El. 216'

Potential location for Containment Failure

Reference: CAR-2165 G-012

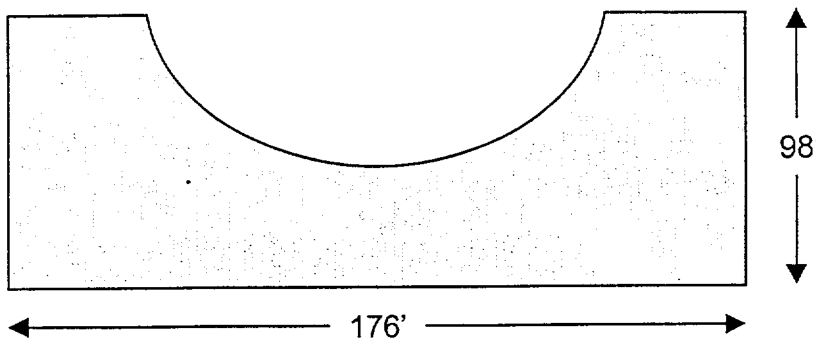


Floor Area	$= 128' x 203' - \frac{1}{2}\frac{\Pi 140^2}{4}$ $= 10{,}590\,ft^2$
Height	= 236 – 216 = 20 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 10,590 x 20 x .90 = 190,620 ft^3
Heat Sink Area	= (2 x (128) + 203) x 20 = 9,180 ft^2

Node 3

RAB El. 236'

Reference: CAR-2165 G-016

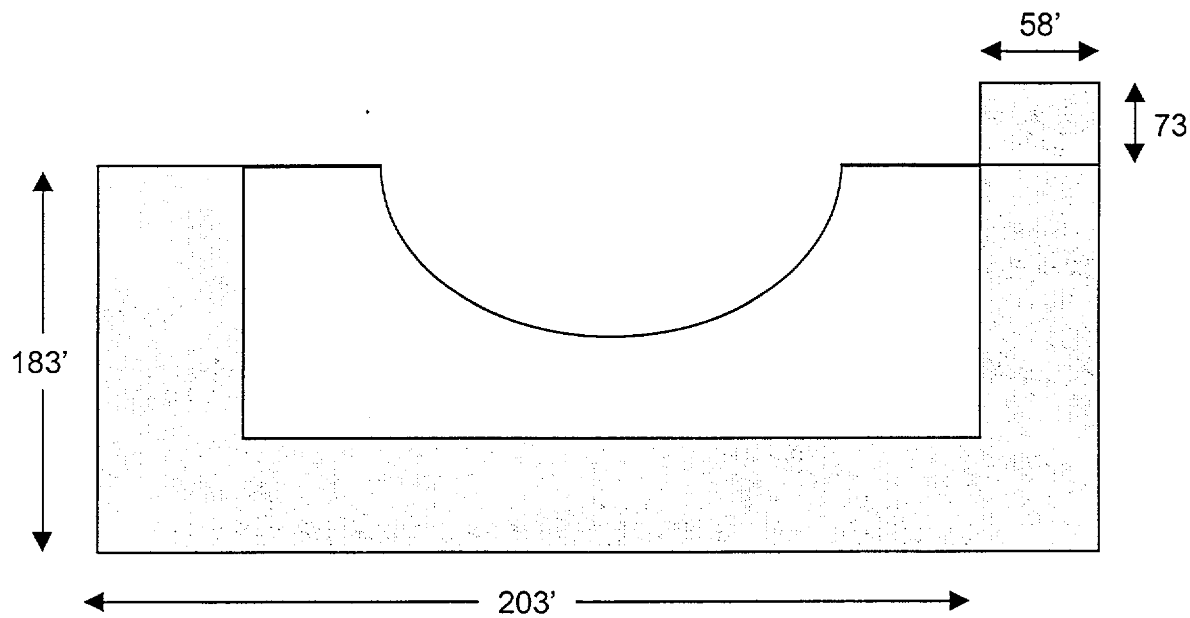


Floor Area	$= 176' \times 98' - \frac{1}{2}\frac{140^2}{4}$ $= 9{,}551\ ft^2$
Height	= 261 – 236 = 25 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 9,551 x 25 x .90 $= 214{,}898\ ft^3$
Heat Sink Area	= (2 x 98 + 176) x 25 $= 9{,}300\ ft^2$

Node 4

RAB El. 236'

Reference: CAR-2165 G-016



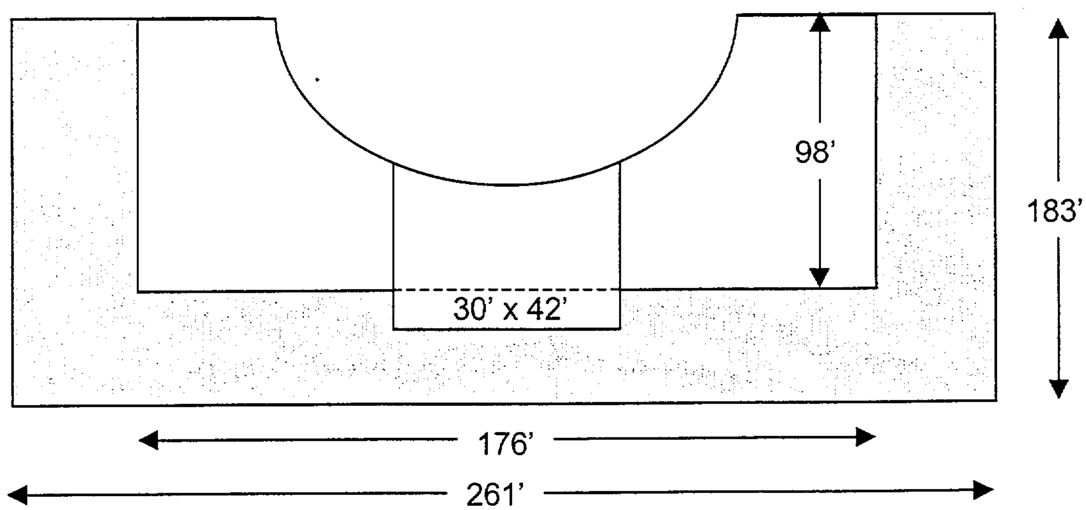
Floor Area	= (203 + 58) x 183 – (176 x 98) + (73 x 58) = 34,749 ft^2
Height	= 261 – 236 = 25 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 34,749 x 25 x .90 = 781,853 ft^3
Heat Sink Area	= (2 x 183 + 203 + 73) x 25 = 16,050 ft^2

Node 5

RAB El. 261' + 286'

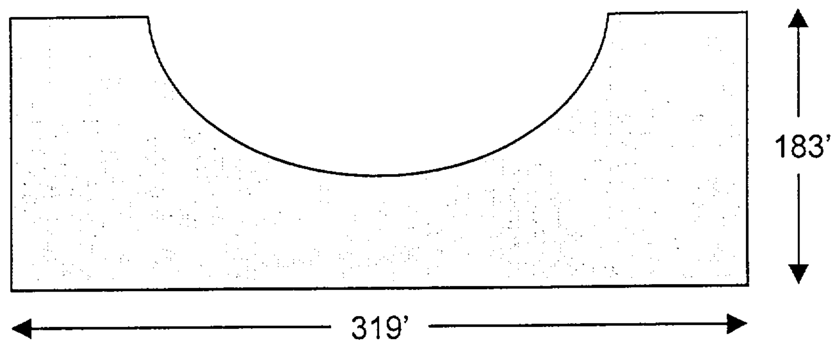
Reference: CAR-2165 G-017, G-018

(El. 261')



Area of ½ Containment Area + Steam Tunnel Area	= 176' x 98' + 30' x 42' = 18,508 ft^2
Floor Area	= 261 x 183 – 18508 = 30,515 ft^2
Height for 261'	= 286 – 261 = 25 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 30,515 x 25 x .90 = 686,588 ft^3

(El. 286')

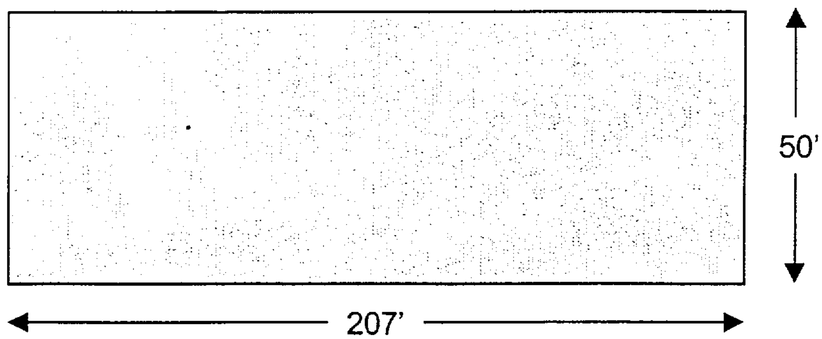


Area	$= 319' \text{x} 183' - \frac{1}{2} \frac{\Pi 140^2}{4}$ $= 50{,}680\,\text{ft}^2$
Height	= 305 – 286 = 19 ft
Volume_{286} (Assume free volume is 90% of total volume to account for structures)	= 50,680 x 19 x .90 = 866,628 ft^3
Total Node 5 Volume	= 1,553,216 ft^3
Floor Area	= 30,515 ft^2 (use smaller value of El. 261' and 286'))
Total Heat Sink Area	= (2 x 183 + 261) x 44 = 25,589 ft^2

Node 6

FHB El. 216' North

Reference: CAR-2165 G-023



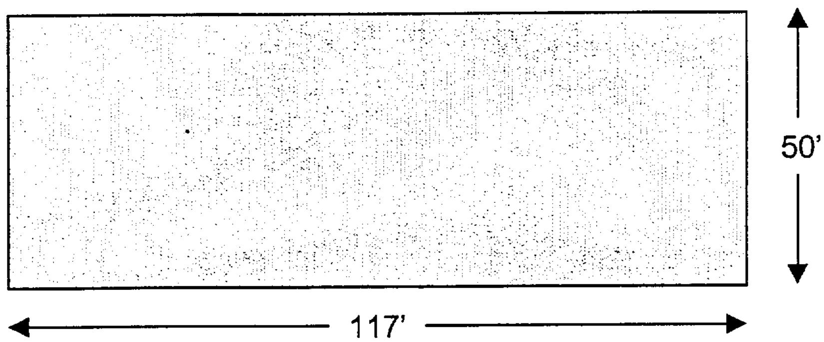
Floor Area	= 207 x 50 = 10,350 ft^2
Height	= 236 – 216 = 20 ft
Volume[1] (Assume free volume is 90% of total volume to account for structures)	= 10,350 x 20 x .90 = 186,300 ft^3
Heat Sink Area	= 2 x (207+ 50) x 20 = 10,280 ft^2

[1] Does not include the added portion of 236' El. North. (This is a small addition and does not affect the calculations.)

Node 7

FHB El. 216' South

Reference: CAR-2165 G-023



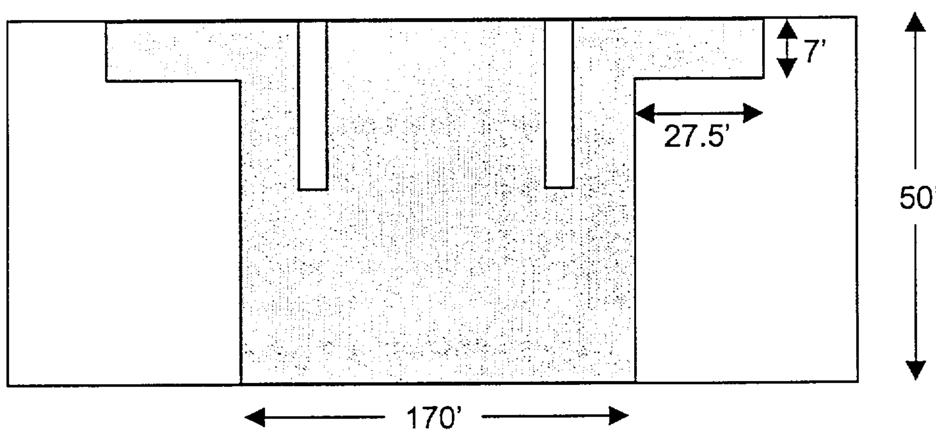
Floor Area	= 117 x 50 = 5,850 ft^2
Height	= 236 – 216 = 20 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 5,850 x 20 x .90 = 105,300 ft^3
Heat Sink Area	= 2 x (117 + 50) x 20 = 6,680 ft^2

Node 8

FHB El. 236'

This node represents the center section on El. 236'. There is also a separate volume on the North end of the FHB that connects to FHB El. 216' North. The North 236' elevation does not communicate with this center region.

Reference: CAR-2165 G-023



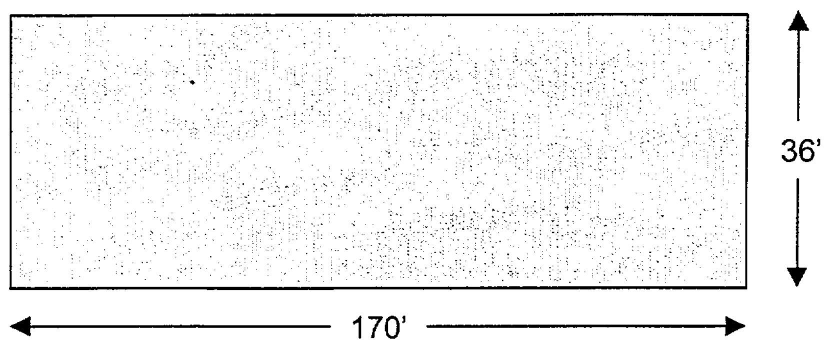
Floor Area	= 170 x 50 + 2 x 27.5 x 7 = 8,885 ft^2
Height	= 261 – 236 = 25 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 8,885 x 25 x .90 = 199,913 ft^3
Heat Sink Area	= 2 x (170 + 27.5 + 50) x 25 = 12,375 ft^2

<u>Node 9</u>

FHB El. 261' + 286'

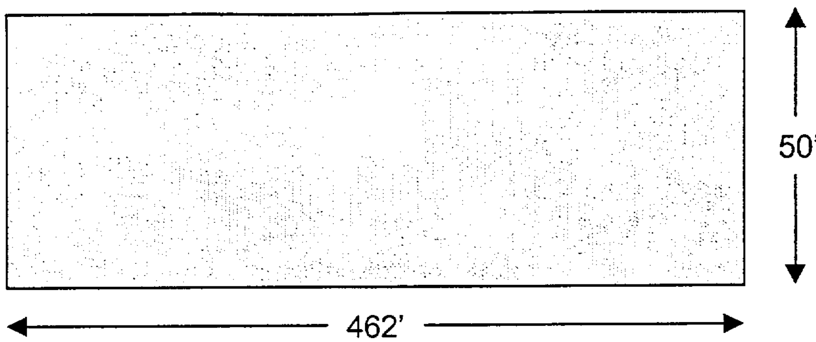
Reference: CAR-2165 G-022

El. 261'



Floor Area	= 36 x 170 = 6,120 ft^2
Height	= 286 – 261 = 25 ft
Volume (Assume free volume is 90% of total volume to account for structures)	= 6,120 x 25 x .90 = 137,700 ft^3

El. 286'



Floor Area	= 462 x 50 = 23,100 ft^2
Height	= 336 – 286 = 50 ft
Volume	= 23,100 x 50 = 1,155,000 ft^3

Total Volume	= 1,292,700 ft^3
Floor Area	= 23,100 ft^2 (use operating deck since it represents most of this volume)
Heat Sink Area	= 2 x (462 + 50) x 50 = 51,200 ft^2

HVAC

The RAB and FHB each have separate normal HVAC systems along with separate emergency exhaust systems. The normal RAB ventilation system shuts down on a safety injection signal and the emergency exhaust starts from the LOCA and LOCA/LOOP programs on the sequencer. The normal FHB ventilation system shuts down on high area radiation levels on the FHB operating deck and the emergency exhaust system starts.

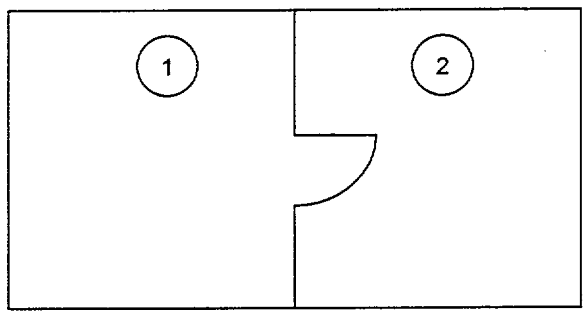
The RAB ventilation system does not communicate with the FHB ventilation system until it reaches the exhaust stack. It is unlikely that one system would backflow into the other since the direction of least resistance is up the stack. The emergency exhaust takes suction from each compartment and discharges it to the stack; therefore, it does not promote mixing between compartments. Also, the normal ventilation systems, i.e., non-safety systems, are powered from non-safety power supplies and will trip and associated dampers will fail closed on the loss of power.

Information provided in the ASHRAE Handbook 1977 Fundamentals classifies ducts as "high pressure" if velocities are greater than 2000 fpm or stack pressures in the duct are between 6 and 10 in. w.g. Assuming a maximum duct pressure of 10 in. w.g. results in a pressure capability of about .4 psid, which is larger than the failure pressure of a door opening away from the jamb. This information would indicate that the doorways, as modeled in the MAAP analysis, would open prior to any failure of the HVAC ductwork. Therefore, the ductwork is not assumed to provide any additional flow paths in the RAB or FHB.

In all of the MAAP analyses the emergency exhaust systems for the RAB and FHB are not assumed to be operating. Overall the operation of these systems should help to reduce the building concentrations of fission products by sweeping out airborne radionuclides.

Definition of Junctions

Connections between the control volumes can be represented as either an open junction or a failure junction. Failure junctions are defined using a failure pressure differential for both the positive and negative flow configurations. For example, the doorway illustrated in the following sketch is modeled to open when the pressure differential between compartment 1 and 2 ($Press_1 - Press_2$) is greater than .25 psid.



For the opposite condition, ($Press_2 - Press_1$), the failure pressure is estimated to be 3 psid.

The MAAP 3.0B auxiliary building model allows us to represent a variety of junction types within the RAB and FHB.

EPRI NP-6586-6, "Evaluation of the Consequences of Containment Bypass Scenarios", investigated the response of secondary containment buildings to severe accident conditions. All secondary containment buildings were categorized relative to their expected behavior and MAAP 3.0B calculations were performed. As part of that evaluation for PWRs, an assessment was made of the pressure capability of a normal doorway. Doors were assumed to open at a pressure differential of 3 psid with the flow in the direction that forced the door into the jamb, and .25 psid with the flow forcing the door away from the jamb. This assumption will be utilized in the Harris RAB/FHB study.

The following describes each junction represented in the Harris MAAP 3.0B model.

Failure #1

Node 1 → Node 2

Doorway into stairwell

Reference: CAR-2165 G-015

The enclosed stairwell connects El.190' and 216'. The limiting door orientation would result in a failure pressure of 3.0 psid from Node 1 to 2, and a failure pressure of .25 in the opposite direction.

The junction is assumed to allow vertical flow from Node 1 to Node 2.

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= 3 psid (Node 1 → Node 2)
Reverse Failure Pressure	= .25 psid (Node 2 → Node 1)

Failure #2

Node 1 → Node 2

Floor hatches

Reference: CAR-2165 G-015

Hatch covers that are simply laid over a floor opening are estimated to lift up due to a pressure differential of .1 psid.

This value was derived by simply estimating the static weight of the hatch cover. Assuming a thickness of ¼" and a steel density of about 500 lb/ft^3 yielded a weight of approximately .1 lbs/in^2. The lifting force is simply the pressure differential to levitate the cover. If flow is in the opposite direction, forcing the cover down, an estimated failure pressure of 2 psid is assumed. This is typical of values used in EPRI-NP-6586.L.

Dimensions	5' x 5'
Area	25 ft^2
Limiting Failure Pressure	= .1 psid (Node 1 → Node 2)
Reverse Failure Pressure	= 2.0 psid (Node 2 → Node 1)

Failure #3

Node 2 → Node 3

Doorway to Stairwell

Reference: CAR-2165 G-015, G-016

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= 3 psid (Node 2 → Node 3)
Reverse Failure Pressure	= 3 psid (Node 3 → Node 2)

A stairwell connects El 216' and El. 236'. On El. 216', a door opens away from the stairwell. On El. 236', the door opens into the stairwell. Using the failure pressures previously defined for doors, failure in either direction is set to 3 psid. This conservatively assumes that all of the resistance to flow occurs through the doorways and not in the stairwell. In either direction of flow, there will be one door opening away from the jamb and one door opening into the jamb. The limiting door opening pressure is assumed to control this junction.

Failure #4

Node 2 → Environment

Railway door from RAB to Waste Processing Building (WPB). The MAAP model assumes flow is directly into environment. The WPB is not currently represented.

Reference: CAR-2165 G-015, G-016

Dimensions	10' x 10'
Area	100 ft^2

Limiting Failure Pressure	= 3 psid (Node 3 → Environment)

Failure #5

Node 3 → Node 4

Doorway connecting inner area of 216' to CCW pumps and heat exchange room.

Reference: CAR-2165 G-016

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= .25 psid (Node 3 → Node 4)
Reverse Failure Pressure	= 3 psid (Node 4 → Node 3)

Failure #6

Node 4 → Environment

Doorway to WPB

Reference: CAR-2165 G-016

Dimensions	10' x 10'
Area	100 ft^2
Limiting Failure Pressure	= 3 psid (Node 4 → Environment)

Failure #7

Node 3 → Node 5

Doorway into Stairwell

Reference: CAR-2165 G-016, G-017

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= .25 psid (Node 3 → Node 5)
Reverse Failure Pressure	= 3 psid (Node 5 → Node 3)

Failure #8

Node 4 → Node 8

Doorway into FHB El. 236'

Reference: CAR-2165 G-016, G-023

Dimensions	10' x 10'
Area	100 ft^2
Limiting Failure Pressure	= 3 psid (Node 4 → Node 8)
Reverse Failure Pressure	= .25 psid (Node 8 → Node 4)

Failure #9

Node 4 → Node 5

Doorway into Stairwell

Reference: CAR-2165 G-016, G-017

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= .25 psid (Node 4 → Node 5)
Reverse Failure Pressure	= 3 psid (Node 5 → Node 4)

Failure #10

Node 5 → Environment

Doorway to WPB

Reference: CAR-2165 G-017

Dimensions	10' x 10'
Area	100 ft^2
Limiting Failure Pressure	= .25 psid (Node 5 → Environment)

Failure #11

Node 8 → Node 9

Hatch Covers

Reference: CAR-2165 G-016, G-017

Dimensions	10' x 10'
Area	2100 ft^2
Limiting Failure Pressure	= .5 psid (Node 8 → Node 9) (The normal lifting pressure is increased to .5 psid to account for screws that hold the cover down).
Reverse Failure Pressure	= 2 psid (Node 9 → Node 8)

Failure #12

Node 9 → Environment

Railway Door at North End of Building at El. 261'. Since FHB 261' and 286' are conservatively combined, the railway door provides a potential release pathway to the environment.

Information included in EPRI NP-6586-6, "Evaluation of the Consequences of Containment Bypass Scenarios", was used for assigning junction failure conditions. Sliding doors were not treated any differently than typical personnel latch doors. The Harris evaluation selected the limiting door failure pressure of .25 psid to represent the sliding door on the FHB 261' North elevation. The large span of this door would make it susceptible to bowing or bending under elevated pressure conditions and it has been assumed to leak or fail at the low end pressure differential of .25 psid. No additional information was found to support a high failure pressure.

Reference: CAR-2165 G-022

Dimensions	10' x 10'
Area	100 ft^2
Failure Pressure for a Sliding Door	Assumed to be .25 psid.

Failure #13

Node 1 → Node 3

Pipe Chase to El. 236'

Reference: CAR-2165 G-015, G-016

Dimensions	5' x 10'
Area	50 ft^2
Open Flow Path	

Failure #14

Node 5 → Node 9

Doorway to FHB El. 261'

Reference: CAR-2165 G-017, G-022

Dimensions	10' x 10'
Area	100 ft^2
Limiting Failure Pressure	= .25 psid (Node 5 → Node 9)
Reverse Failure Pressure	= 3 psid (Node 9 → Node 5)

Failure #15

Node 7 → Node 8

Hatch Cover

Reference: CAR-2165 G-023

Dimensions	10' x 10'
Area	100 ft^2
Limiting Failure Pressure	= .1 psid (Node 7 → Node 8)
Reverse Failure Pressure	= 2 psid (Node 8 → Node 7)

Failure #16

Node 6 → Node 8

Hatch Cover (Locked Down)

Reference: CAR-2165 G-023

Dimensions	10' x 10'
Area	100 ft^2
Limiting Failure Pressure	= 2 psid (Node 6 → Node 8)
Reverse Failure Pressure	= 2 psid (Node 8 → Node 6)

Hatch cover is locked in place as shown in photos taken during walkdown. Failure pressure assumed to be 2 psid.

Failure #17

Node 6 → Node 9

Doorway to Stairwell

Reference: CAR-2165 G-022

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= .25 psid (Node 6 → Node 9)
Reverse Failure Pressure	= 3 psid (Node 9 → Node 6)

Junction #18

Node 1 → Node 2

This junction represents open gaps around penetrations.

Reference: Walkdown

Dimensions	6' x 6'
Area	2 ft^2
Open Pathway	

Junction #19

Node 8 → Node 9

This junction represents various open gaps around penetrations.

Reference: Walkdown

Dimensions	5' x 5'
Area	.5 ft^2

Junction #20

Node 7 → Node 8

This junction is used to represent open gaps around penetrations.

Reference: Walkdown

Dimensions	5' x 5'
Area	.5 ft^2
Open Flowpath	

Junction #21

Node 6 → Node 8

This junction is used to represent open gaps around penetrations.

Reference: Walkdown

Dimensions	5' x 5'
Area	.5 ft^2
Open Flowpath	

Failure #22

Node 2 → Node 7

Doorway

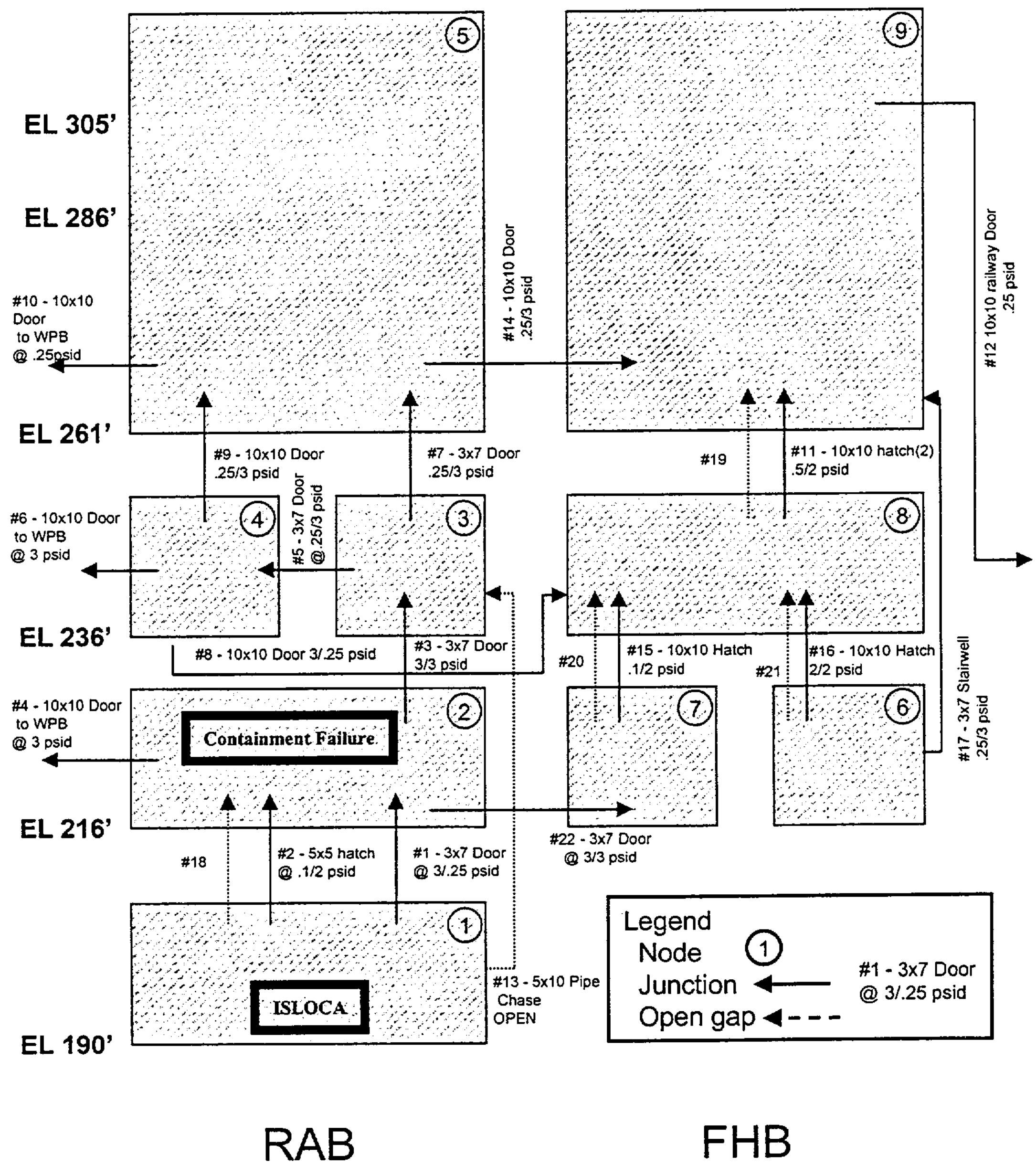
Reference: CAR-2165 G-016, G-017

Dimensions	3' x 7'
Area	21 ft^2
Limiting Failure Pressure	= 3 psid (Node 2 → Node 7)
Reverse Failure Pressure	= .25 psid (Node 7 → Node 2)

Summary

Figure E.2-1 illustrates the Node and Junction modeling for the Harris RAB and FHB representation.

Figure E.2-1 MAAP Nodalization



E.3 DETAILED MAAP RESULTS

E.3.1 Interfacing System LOCA

This scenario is initiated by a 12" break in the cold leg with release into the RHR pump room located on the 190' elevation of the RAB. Table E.3.1-1 provides a brief time line for this accident scenario.

Table E.3.1-1 – ISLOCA Timeline

Time (hr)	Event Description
0	12" break in cold leg releasing to 190' of RAB Reactor Scram HPI/LPI Failure Main FW/Aux FW Failure Pressurizer sprays/heater failed
.36	Core Uncovers
1.29	Vessel Failure

As the primary system begins to discharge into the RHR pump room (RAB Node 1), the RAB pressure begins to increase resulting in various doors and hatches opening to allow flow into adjacent RAB and FHB compartments. Figure E.3.1-1 shows the MAAP nodalization drawing with all active flow paths identified. The direction of the flow paths for positive flow only is shown with an arrow. Once a junction has failed open, flow is able to occur in both the positive and negative direction. In addition to unidirectional flow, each junction has the potential for counter-current flow under the appropriate circumstances. Also identified on Figure E.3.1-1 are leakage pathways represented in the model.

Following the initiation of the break in the primary system, RAB 190' begins to pressurize rapidly. As seen in Figure E.3.1-1, the door (Junction #1) leading into the stairwell opens allowing flow into the RAB 216' elevation. In addition, the following junctions fail open:

- Junction #2: Hatch cover on the floor of RAB 216'
- Junction # 3: Door into stairwell from RAB 216' up to RAB 236'
- Junction #4: Door on RAB 216' leading into the Waste Processing Building (WPB)
- Junction #5: Door connecting the interior region on RAB 236' to the CCW pump area on RAB 236'
- Junction #7: Door into stairwell from RAB 236' Node 3 up to RAB 261'
- Junction #9: Door into stairwell from RAB 236' Node 4 up to RAB 261'
- Junction #10: Door on RAB 261' leading into the WPB
- Junction #11: Hatch cover in floor connecting FHB 236' to FHB 261'
- Junction #12: Railway door to outside at FHB 261'
- Junction #14: Door connecting RAB 261' to FHB 261'
- Junction #15: Hatch cover in floor of FHB 236' connecting to FHB 216' South (Node 7)
- Junction #22: Door connecting RAB 216' to FHB 216' South

The flow of high temperature gas into RAB and FHB compartments may impact the success of systems required for cooling and makeup to the SFP. Table E.3.1-2 provides the peak temperatures calculated at the various elevations:

Table E.3.1-2 ISLOCA: Peak Compartment Temperatures

Node #	**Location**	**Key Equipment**	**Peak Temperature (°F)**
4	RAB 236'	CCW Pumps and Heat Exchangers	200
6	FHB 216' North	Purification Pumps for C/D Pools	80
7	FHB 216' South	Purification Pumps for A/B Pools	280
8	FHB 236'	Fuel Pool Cooling and Skimmer Pumps and Local Controls	250
9	FHB 286'	Local Controls for purification and skimmer pumps and various makeup sources	170

Given the active flow paths illustrated in Figure E.3.1-1, high radiation is expected in several of the RAB and FHB areas once the core has uncovered and begun to heat up. Immediately following the release of radionuclides into the RAB and FHB, all of the elevations are expected to experience high dose levels with the exception of the FHB 236' and 216' North areas. The FHB 236' North area is assumed part of Node 6 and is separate from the central area represented by Node 8. This area does not see a direct flow of gas and radionuclides and can be accessed from outside through a separate entrance on the 236' elevation. As identified in the walkdown, entry from the FHB 236' North elevation into the FHB 216' North elevation can be achieved using the ladder mounted on the wall. This avoids entry into the stairwell which could have higher airborne dose levels due to it being open to the operating deck (FHB 286').

The fission product mass in each of the key areas was provided to CP&L's Radiation Protection Department to assess the actual dose levels in the FHB. These analyses have been performed and are included in a separate document.

The MAAP input file, event summary file, and detailed plots showing the response of the primary system, containment, and adjacent buildings are included at the end of the appendix.

E.3.2 Steam Generator Tube Rupture

This scenario is initiated by a steam generator tube rupture with subsequent failure of the faulted steam generator relief valve in the open position. This provides a direct pathway from the primary system to outside of the containment and adjacent buildings. High pressure injection (HPI) is also assumed to fail in this event leading to core uncovery and eventual core damage. Table E.3.2-1 provides a time line of key events for this accident scenario.

Table E.3.2-1 – SGTR Timeline

Time (hr)	Event Description
0	SGTR Reactor Scram Stuck open steam generator relief valve HPI Failure Pressurizer sprays/heater failed
3.6	Core Uncovers
6.8	Vessel Failure

Following reactor scram, the steam generator pressure operated relief valve (PORV) cycles open and closed for a period out to about 90 seconds into the event. At 126 seconds, a steam generator safety valve opens and is assumed to be stuck in that position. This allows for a direct pathway from the primary system directly out of the containment to the environment. The main coolant pumps are tripped off as a result of increased voids in the primary system at about 33 minutes into the event. Due to boil-off, the broken steam generator dries out at about 2.1 hours. At this time the broken tube is uncovered and any fission products released are transported directly through the open steam generator safety valve. According to the procedures, the operator is assumed to isolate feedwater and close the PORVs on the broken steam generator at 1 minute into the accident.

The fission product release fractions to the environment were provided to CP&L's Radiation Protection Department to assess the actual dose levels in surrounding site in order to assess the viability of moving personal for potential mitigation actions. These analyses have been performed and are included in a separate document.

The MAAP input file, event summary file, and detailed plots showing the response of the primary system, containment, and adjacent buildings are included at the end of the appendix.

E.3.3 Containment isolation Failure

This scenario is initiated by closure of the main steam isolation valves with subsequent system failures resulting in core damage. It is also assumed that there is failure to isolate containment resulting in a 5-inch diameter opening from containment into the RAB 236"elevation. Table E.3.3-1 provides a brief time line for this accident scenario.

Table E.3.3-1 – Containment isolation Failure Timeline

Time (hr)	Event Description
0	Reactor Scram Main coolant pumps off HPI and low pressure injection (LPI) failed 5" containment isolation failure Containment sprays and fan coolers off Pressurizer sprays/heater failed
4.0	AFW off
5.2	Core Uncovers
8.0	Vessel Failure

The primary system pressure is approximately 600 psia at the time of vessel failure and the containment responds by an indicated pressure rise of about 32 psia.

The following junctions are observed to fail open:

- Junction #5: Door connecting the interior region on RAB 236' to the CCW pump area on RAB 236'
- Junction #7: Door into stairwell from RAB 236' Node 3 up to RAB 261'
- Junction #10: Door on RAB 261' leading into the WPB

Reviewing the details of the flow patterns after the initial failures shows that the gas released into the RAB 236' elevation is transported down into the RAB 190' elevation through the open gaps and then up to the RAB 236' elevation through the pipe chase. The dynamic response for this scenario is smaller than for the ISLOCA scenario and, as a result, fewer door junctions fail in the RAB.

Flow from the RAB to the FHB does not occur, leaving the entire FHB unaffected.

The flow of high temperature gas into RAB compartments may impact the success of systems required for cooling and makeup to the SFP. Table E.3.3-2 provides the peak temperatures calculated at the various elevations:

Table E.3.3-2 Containment isolation Failure: Peak Compartment Temperatures

Node #	Location	Key Equipment	Peak Temperature (°F)
4	RAB 236'	CCW Pumps and Heat Exchangers	170
6	FHB 216' North	Purification Pumps for C/D Pools	80
7	FHB 216' South	Purification Pumps for A/B Pools	80
8	FHB 236'	Fuel Pool Cooling and Skimmer Pumps and Local Controls	80
9	FHB 286'	Local Controls for Purification and Skimmer Pumps and various makeup sources	80

Given the active flow paths illustrated in Figure E.3.3-1, high radiation is expected in several of the RAB areas once the core has uncovered and begun to heat up. Immediately following the release of radionuclides into the RAB, all of the elevations in the RAB are expected to experience high dose levels. All elevations of the FHB are expected to be generally unaffected by the accident conditions.

The fission product mass in each of the key areas was provided to CP&L's Radiation Protection Department to assess the actual dose levels in the FHB. These analyses have been performed and are included in a separate document.

The MAAP input file, event summary file, and detailed plots showing the response of the primary system, containment, and adjacent buildings are included at the end of the appendix.

E.3.4 Early Containment Failure

This scenario is initiated by closure of the main steam isolation valves with subsequent system failures resulting in core damage. It is also assumed that the containment fails as a result of vessel breach. Table E.3.4-1 provides a brief time line for this accident scenario.

Table E.3.4-1 – Early Containment Failure Timeline

Time (hr)	Event Description
0	Reactor Scram Main coolant pumps off Charging pumps failed LPI failed Containment sprays and fan coolers off Pressurizer sprays/heater failed Makeup and letdown failed
2.2	Core Uncovers
3.6	Vessel Failure and Containment Failure

The primary system pressure is approximately 2000 psia at the time of vessel failure. The assumed containment failure results in opening of junctions in the RAB.

The following junctions are observed to fail open for the early containment failure case:

- Junction #5: Door connecting the interior region on RAB 236' to the CCW pump area on RAB 236'

- Junction #7: Door into stairwell from RAB 236' Node 3 up to RAB 261'
- Junction #10: Door on RAB 261' leading into the WPB
- Junction #14: Door connecting RAB 261' to FHB 261'

Reviewing the details of the flow patterns after the initial failures shows that the gas released into the RAB 236' elevation is transported down into the RAB 190' elevation through the open gaps and then up to the RAB 236' elevation through the pipe chase. The dynamic response for this scenario is less severe than for the ISLOCA scenario and, as a result, fewer junctions fail in the RAB.

Flow from the RAB to the FHB occurs through the doorway on the 261' elevation, leaving the lower FHB elevations generally unaffected.

The flow of high temperature gas into RAB and FHB compartments may impact the success of systems required for cooling and makeup to the SFP. Table E.3.4-2 provides the peak temperatures calculated at the various elevations:

Table E.3.4-2 Early Containment Failure: Peak Compartment Temperatures

Node #	Location	Key Equipment	Peak Temperature (°F)
4	RAB 236'	CCW Pumps and Heat Exchangers	190
6	FHB 216' South	Purification Pumps for A/B Pools	80
7	FHB 216' North	Purification Pumps for C/D Pools	80
8	FHB 236'	Fuel Pool Cooling and Skimmer Pumps and Local Controls	80
9	FHB 286'	Local Controls for Purification and Skimmer Pumps and various makeup sources	150

Given the active flow paths illustrated in Figure E.3.4-1, high radiation is expected in several of the RAB and FHB areas once the core has uncovered and begun to heat up. Immediately following the release of radionuclides into the RAB, all of the elevations in the RAB are expected to experience high dose levels. Only the operating deck of the FHB is expected to see increased dose levels for this scenario. The 236' and 216' elevations of the FHB are expected to be generally unaffected by the accident conditions.

The fission product mass in each of the key areas was provided to CP&L's Radiation Protection Department to assess the actual dose levels in the FHB. These analyses have been performed and are included in a separate document.

The MAAP input file, event summary file, and detailed plots showing the response of the primary system, containment, and adjacent buildings are included at the end of the appendix.

E.3.5 Late Containment Failure

This scenario is initiated by closure of the main steam isolation valves with subsequent system failures resulting in core damage. In this case the containment pressure slowly increases after vessel failure until reaching the ultimate capacity at about 2 days into the event. Table E.3.5-1 provides a brief time line for this accident scenario.

Table E.3.5-1 – Late Containment Failure Timeline

Time (hr)	Event Description
0	Reactor Scram Main coolant pumps off Main feed water off LPI failed Containment sprays and fan coolers off Pressurizer sprays/heater failed Makeup and letdown failed
8.1	HPI fails on low RWST level
9.6	Core Uncovers
12.0	Vessel Failure
42.9	Containment Failure

Containment failure occurs when the pressure reaches a value of 145 psia. The following junctions in the RAB and FHB are observed to fail open:

- Junction # 1: Door into stairwell from RAB 216' up to RAB 236'
- Junction #5: Door connecting the interior region on RAB 236' to the CCW pump area on RAB 236'
- Junction #7: Door into stairwell from RAB 236' Node 3 up to RAB 261'

- Junction #9: Door into stairwell from RAB 236' Node 4 up to RAB 261'
- Junction #10: Door on RAB 261' leading into the WPB
- Junction #14: Door connecting RAB 261' to FHB 261'

Reviewing the details of the flow patterns after the initial failures shows that the gas released into the RAB 216' elevation is transported down into the RAB 190' elevation through the open gaps and stairwell and then up to the RAB 236' elevation through the pipe chase.

Flow from the RAB to the FHB occurs through the doorway on the 261' elevation, leaving the lower FHB elevations generally unaffected.

The flow of high temperature gas into RAB and FHB compartments may impact the success of systems required for cooling and makeup to the SFP. Table E.3.5-2 provides the peak temperatures calculated at the various elevations:

Table E.3.5-2 Late Containment Failure: Peak Compartment Temperatures

Node #	Location	Key Equipment	Peak Temperature (°F)
4	RAB 236'	CCW Pumps and Heat Exchangers	240
6	FHB 216' South	Purification Pumps for A/B Pools	80
7	FHB 216' North	Purification Pumps for C/D Pools	80
8	FHB 236'	Fuel Pool Cooling and Skimmer Pumps and Local Controls	80
9	FHB 286'	Local Controls for Purification and Skimmer Pumps and various makeup sources	180

Given the active flow paths illustrated in Figure E.3.5-1, high radiation is expected in several of the RAB and FHB areas once the core has uncovered and begun to heat up. Immediately following the release of radionuclides into the RAB, all of the elevations in the RAB are expected to experience high dose levels. Only the operating deck of the FHB is expected to see increased dose levels for this scenario. The 236' and 216' elevations of the FHB are expected to be generally unaffected by the accident conditions.

The fission product mass in each of the key areas was provided to CP&L's Radiation Protection Department to assess the actual dose levels in the FHB. These analyses have been performed and are included in a separate document.

The MAAP input file, event summary file, and detailed plots showing the response of the primary system, containment, and adjacent buildings are included at the end of the appendix.

E.3.6 Spent Fuel Pool Boiling calculation

An additional MAAP calculation was performed to investigate the temperature response of the FHB to boiling in the SFPs. MAAP 3.0B allows the user to input mass and energy flows into one of the RAB/FHB nodes without exercising the primary system and containment models in MAAP.

To bound the problem, the maximum spent fuel pool heat loads are used:

Pools A/B	25,000,000 BTU/hr
Pools C/D	15,661,901 BTU/hr
Total	40,661,901 BTU/hr

All of this heat is assumed to result in boiling of the pool water using saturated conditions at 1 atmosphere.

Figure E.3.6-1 through E.3.6-4 show the gas temperatures in the FHB as a result of boiling in the pools. Note that only the operating deck (El. 286') heats up significantly, with the lower elevations remaining generally unaffected by the boiling. Junction #12, the railway door, opens up as a result of the pressure increase and provides a release pathway for the steam.

The conclusion from this calculation is that even with boiling in the SFPs, access to the lower elevations should remain possible.

Figure E.3.1-1 – ISLOCA: Active Flow Paths

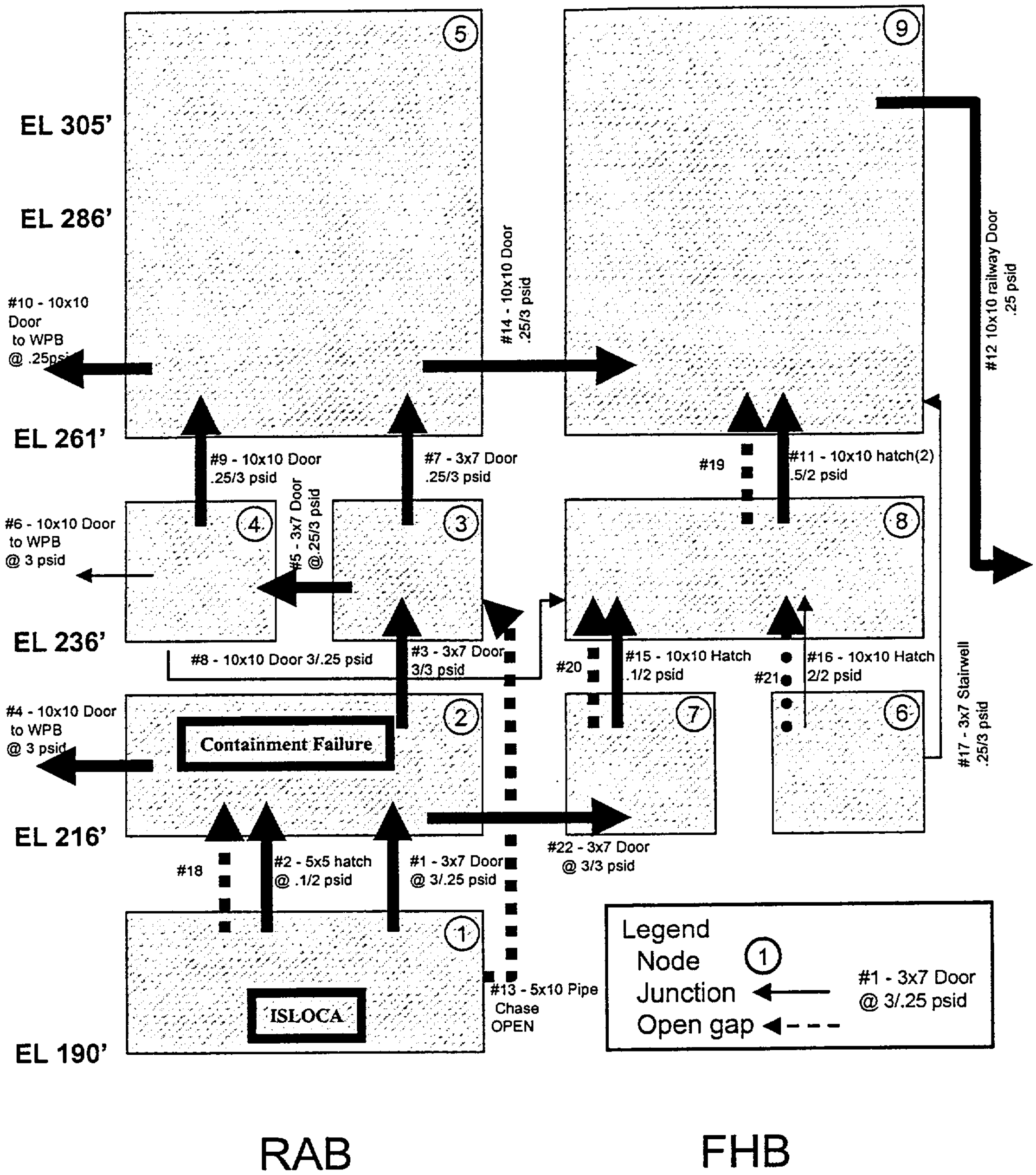


Figure E.3.3-1 – Containment isolation Failure: Active Flow Paths

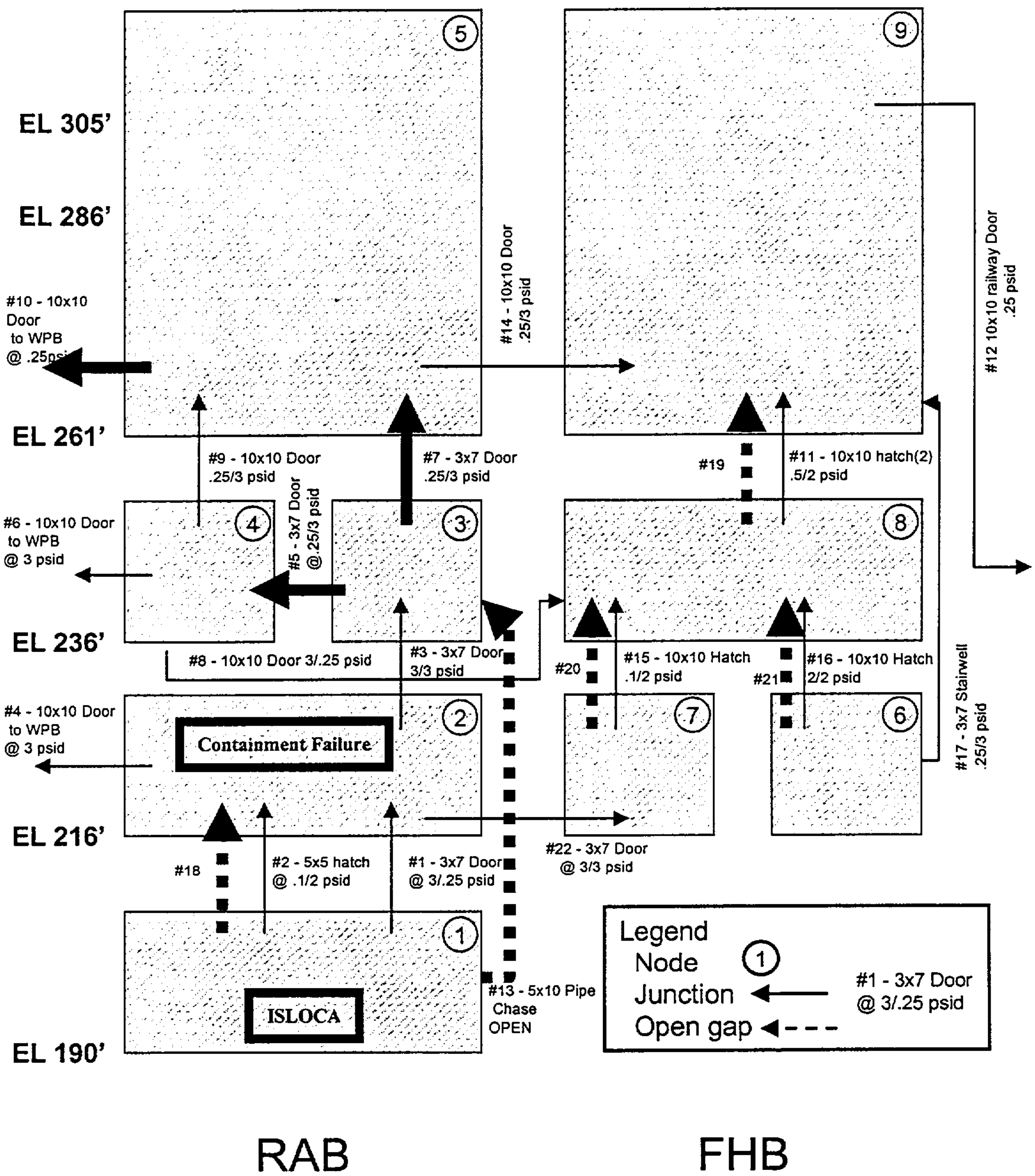


Figure E.3.4-1 – Early Containment Failure: Active Flow Paths

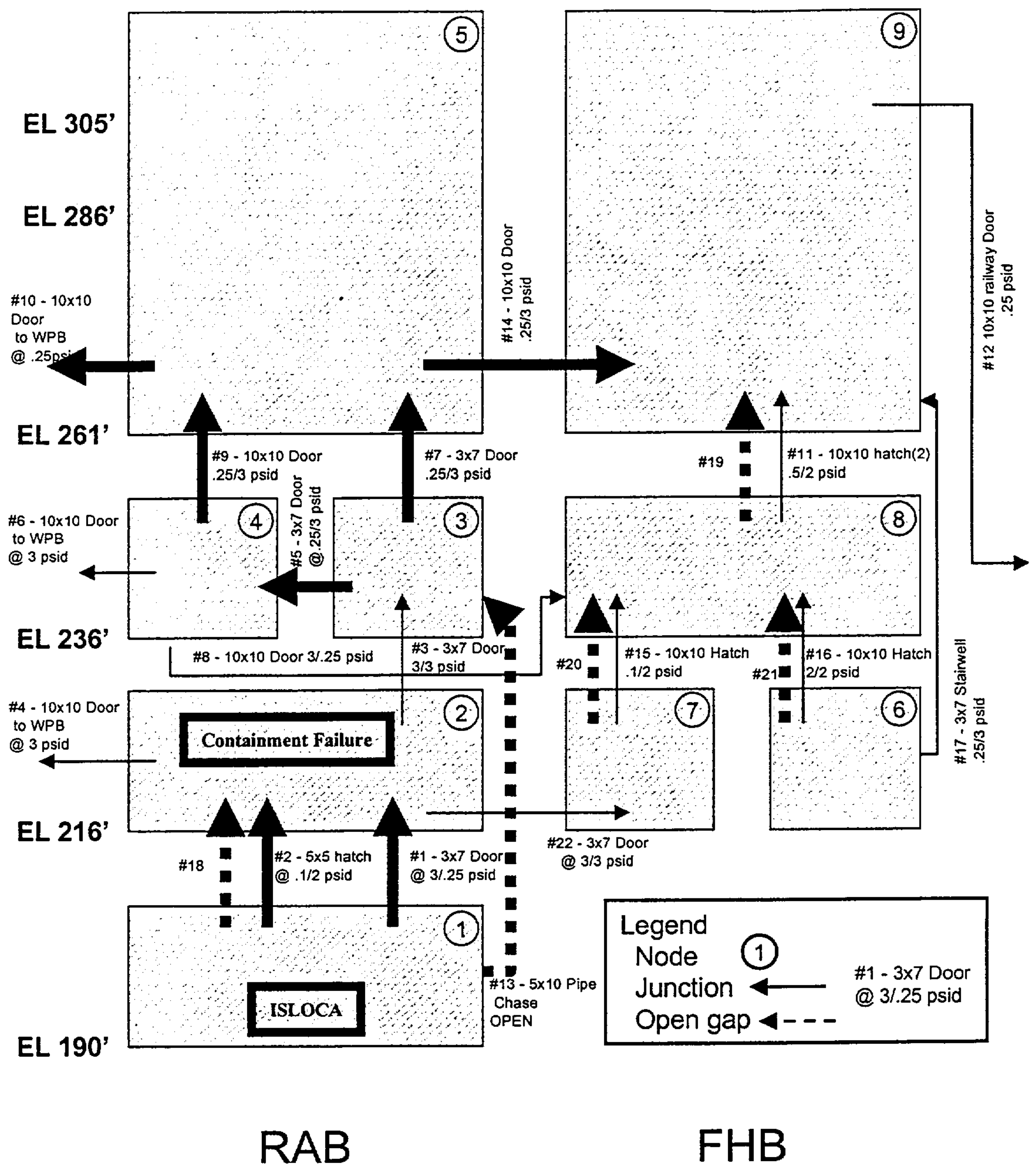


Figure E.3.5-1 – Late Containment Failure: Active Flow Paths

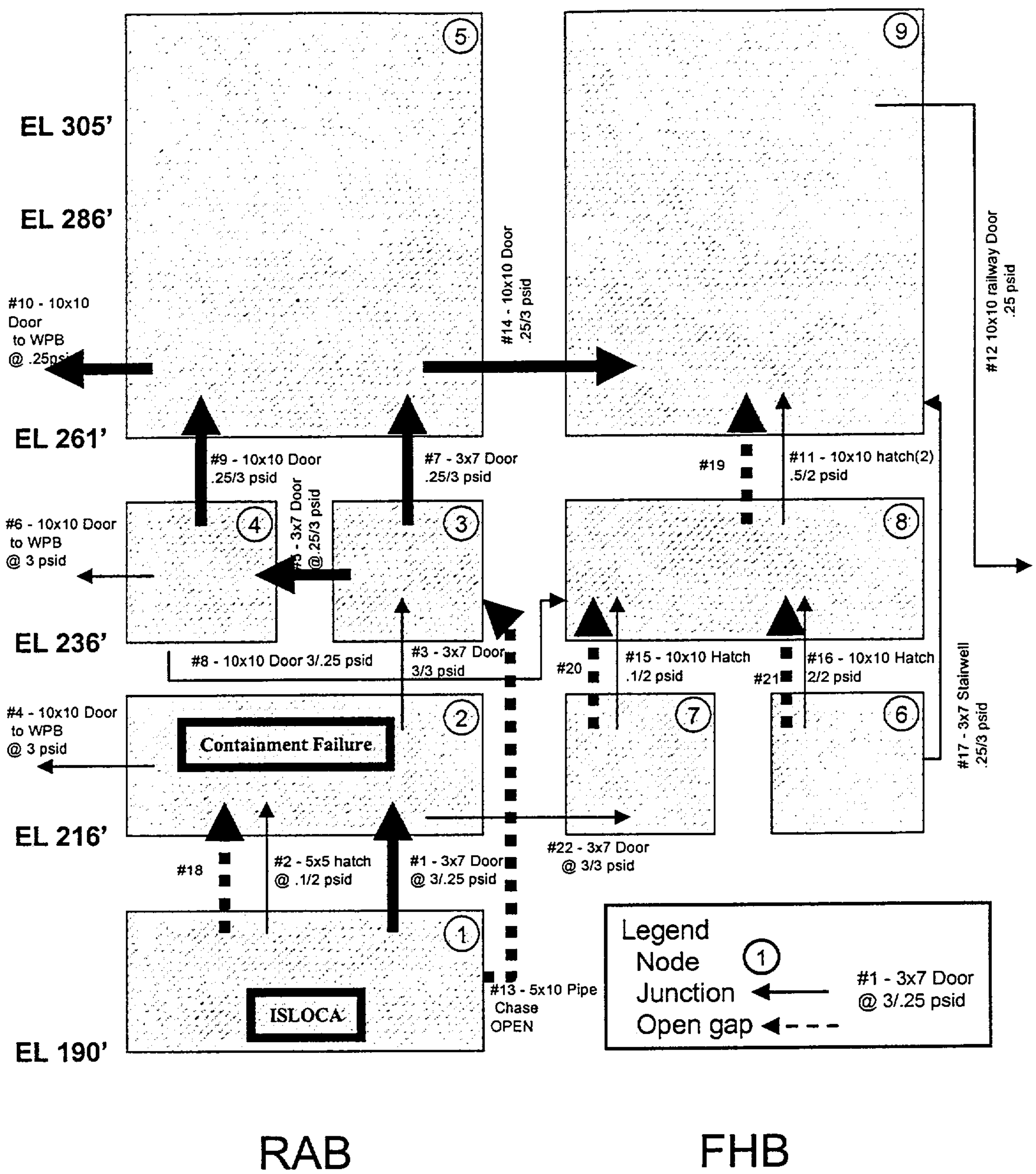


Figure E.3.6-1 – Temperature (°F) – FHB El. 286'

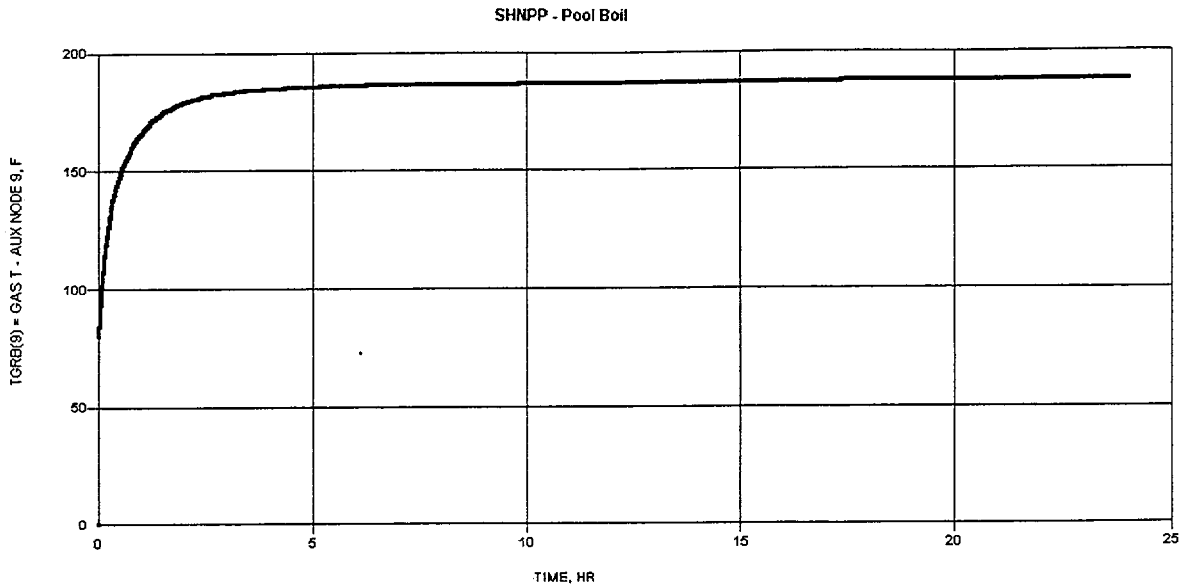


Figure E.3.6-2 – Temperature (°F) – FHB El. 236'

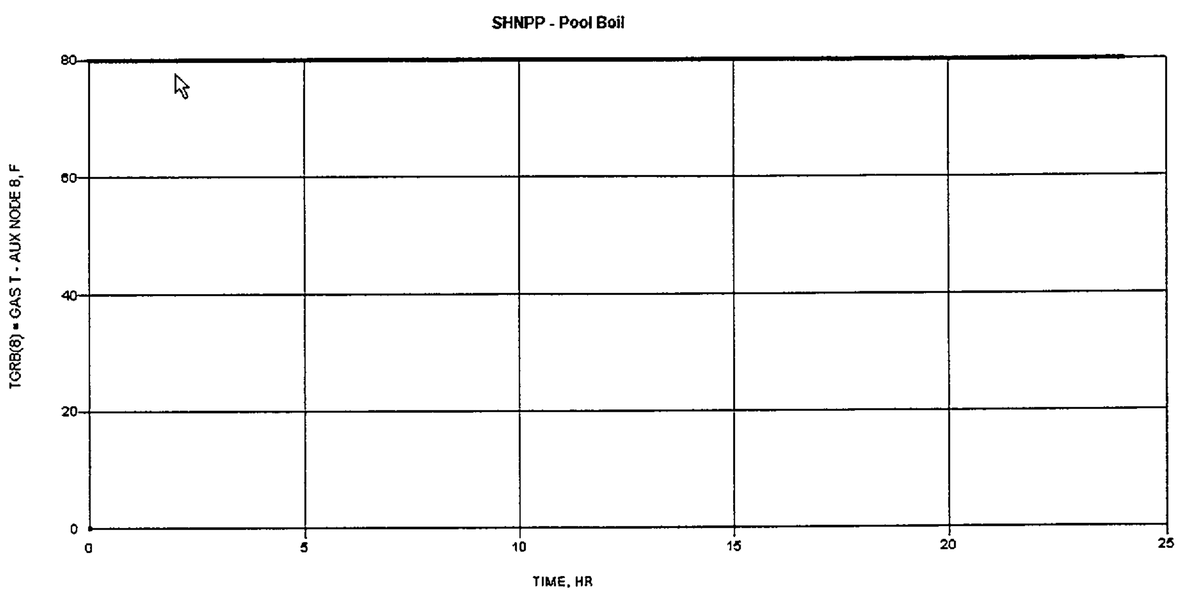


Figure E.3.6-3 – Temperature (°F) – FHB El. 216' South

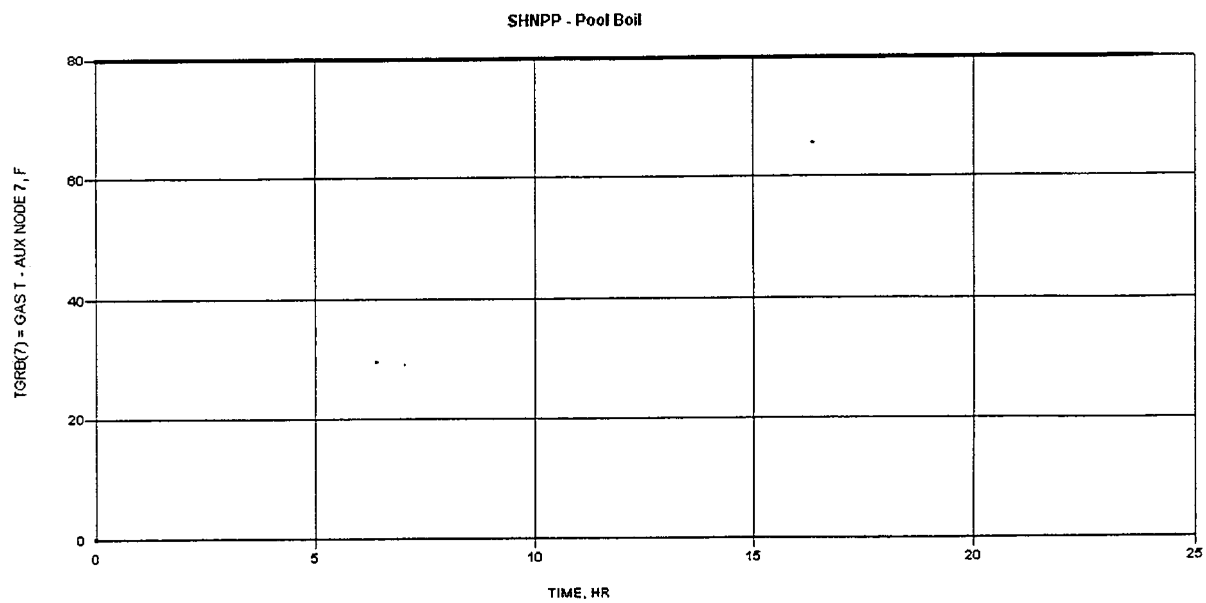
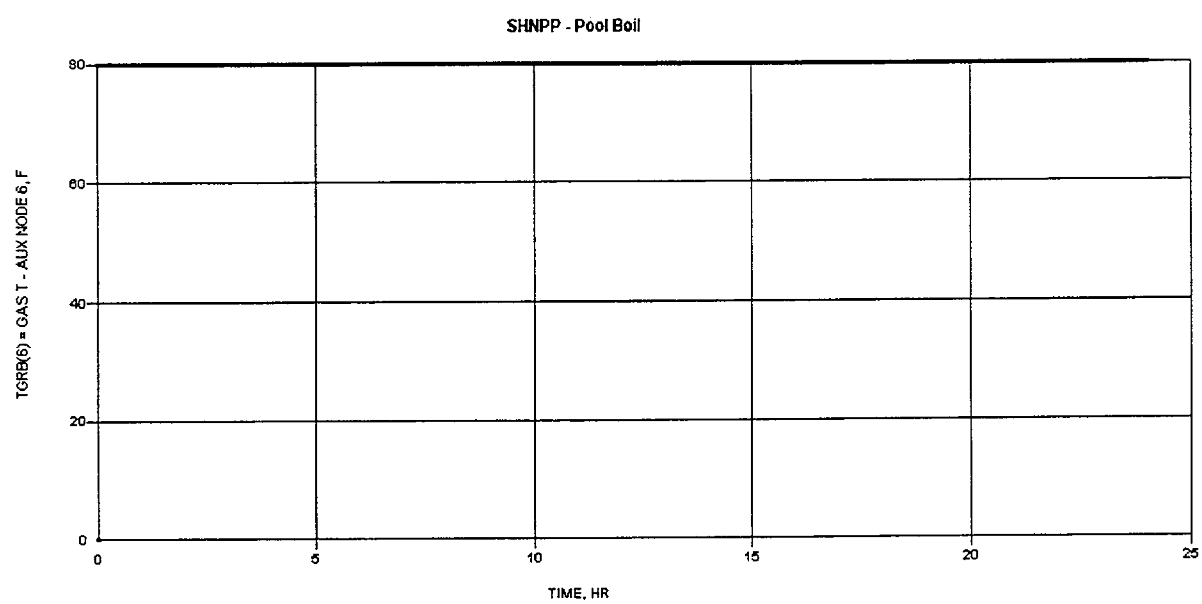


Figure E.3.6-4 – Temperature (°F) – FHB El. 216' North



E.4 SUMMARY

The response of the plant and the equipment following different accident sequences can be markedly different due to the significantly different environmental conditions that can be caused by severe accidents being assessed as part of the ASLB order.

Therefore, as part of the PSA to address the postulated sequence of events, a number of detailed deterministic evaluations have been performed using MAAP 3.0B for the assessment of the following:

- Access to compartments
- Equipment operability in various compartments.

Figure E.4-1 shows the important locations within the Reactor Auxiliary Building (RAB) and Fuel Handling Building (FHB). The results indicate the following for:

- Accessibility (see Table E.4-1)
- Pump operability (see Table E.4-2)

Table E.4-1
Summary of Accessibility Limitations as a Function of Severe Accident Conditions Due to Radiation

Containment Failure Mode	Location				
	RAB	FHB El. 286' (& 261')	FHB El. 236'	FHB El. 216' N (& 236' N)	FHB El. 216' S
ISLOCA	X	X	X	A	X
SGTR	X[1]/A[2]	X[1]/A[2]	A[2]	A[2]	X[3]/A[2]
Containment Isolation Failure	X	A	A	A	X[3]
Early Containment Failure	X	X	A	A	X[3]
Late Containment Failure	X[1]/A[2]	X[1]/A[2]	A	A	X[1,3]/A[2]
Spent Fuel Pool Boiling	A	X[4]	A	A	A

LEGEND
X - Means that for the indicated core damage and containment failure mode, the location is NOT accessible for personnel.
A - Accessible

[1] The inaccessibility is for times AFTER containment failure.
[2] Areas are accessible for the time before containment failure.
[3] Requires access to RAB 216' El. Therefore, access is not available.
[4] The inaccessibility due to high temperatures is for times AFTER the onset of pool boiling.

Table E.4-2

SUMMARY OF EQUIPMENT SURVIVABILITY AS A FUNCTION OF SEVERE ACCIDENT CONDITIONS

Containment Failure Mode	Locations with Potential Equipment Failures				
	RAB	FHB El. 286' (and 261')	FHB El. 236'	FHB El. 216' N (and 236' N)	FHB El. 216' S
ISLOCA	X	X	X	A	X
SGTR	A/X	A/X	A	A	A
Containment Isolation Failure	X	X	A	A	A
Early Containment Failure	X	X	A	A	A
Late Containment Failure	A/X	A/X	A	A	A

LEGEND

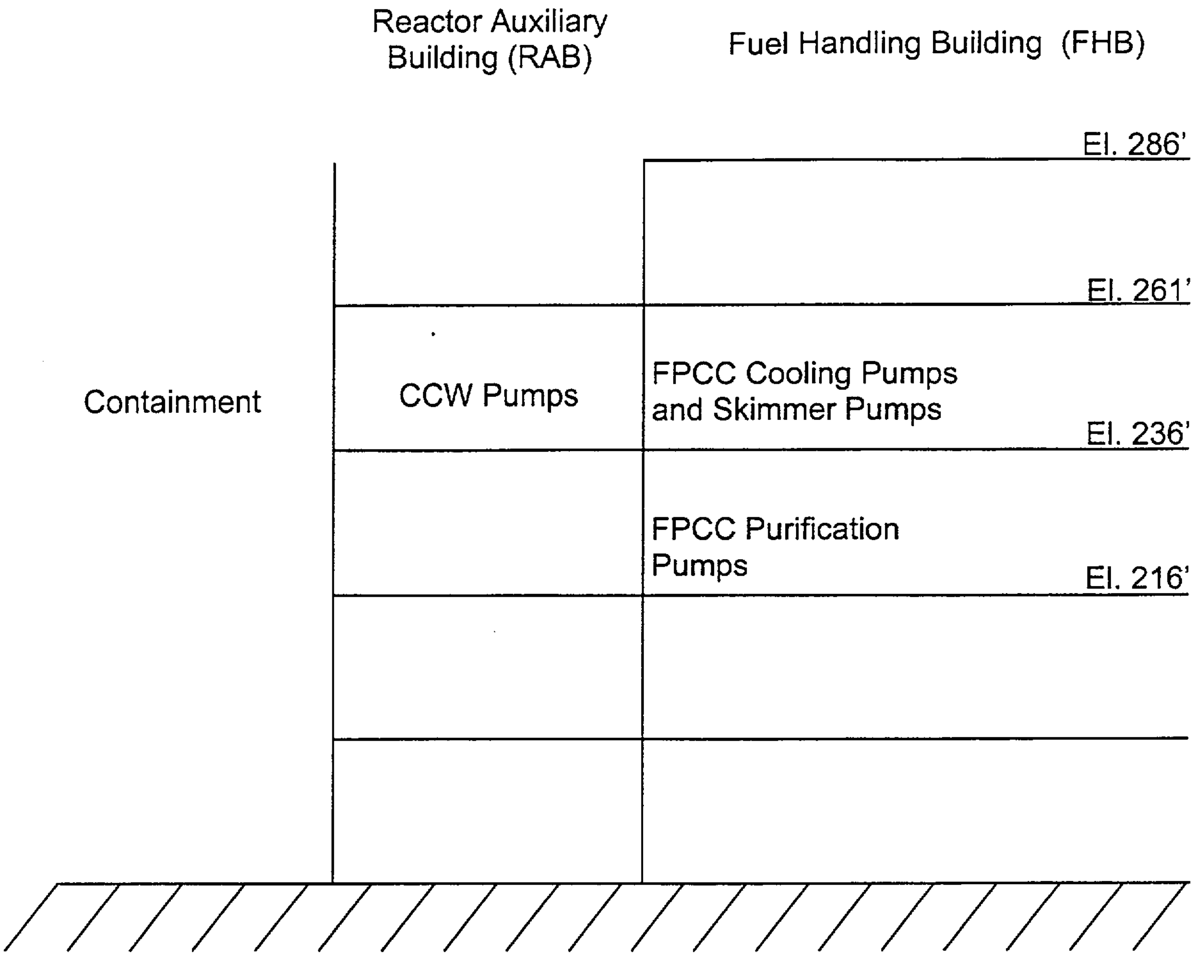
A - Pumps are considered to have survived the environment.

X - Means that for the indicated core damage and containment failure mode pumps in the location are NOT considered to survive the environment.

A/X - Pumps assumed to operate successfully before containment failure. (See Section 2.4 for containment failure times as a function of accident type.)

Figure E.4-1

Simplified Drawing to Show Critical Locations



E.5 REFERENCES

E-1 Modular Accident Analysis Program (MAAP) for PWR Revision 20.

E-2 MAAP –3.0B – Modular Accident Analysis Program for LWR Power Plants EPRI NP-7071-CCML November 1990 (Users Manual)

E-3 Shearon Harris Individual Plant Examination

MAAP RESULTS

- Input File
- Event Summaries
- Plots

```
C TEST
TITLE
        SHNPP - ISLOCA RHR Room El 190'
        END TITLE

ATTACH ATTACH.SAM

PARAMETER CHANGES
C  ASSUME A 12" DIAMETER RHR PIPE BREAK
        BREAK AREA 0.785 FT**2
        TDMAX    5   SECONDS
        END OF PARAMETER CHANGES AND NOLIST

NOT A RESTART

PRINT TIME 5 HOURS

FINAL TIME 10.0

PARALLEL

WHEN BEGIN
     V SEQUENCE ON
     SCRAM ON
     BREAK ON
     HPI OFF
     LPI OFF
     CHARGING PUMPS OFF
     MAIN FED WATER PUMPS OFF
     AUX FEEDWATER PUMPS OFF
     PZR HEATERS OFF
     PZR SPRAYS OFF
     END

WHEN PBS > $PSGSVL$
     LABEL: SG SRV STUCK OPEN
     IEVNT(239) ON
     END

WHEN ZWRWST < 9.31 FT
     LPI ON
     RECIRCULATION MODE ON
     FND

INTERVENTION 49
WHEN RPV FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 50
WHEN CONTMT FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 51
```

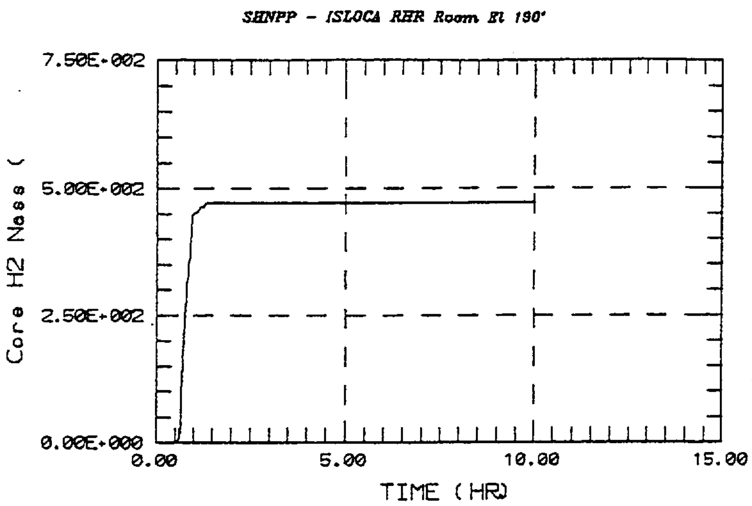
```
WHEN CORE UNCOVERED IS TRUE
LET CORE UNC TIME = TIME
END

INTERVENTION 52
WHEN HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
  OR HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
LET MELT ONSET TIME = TIME
END

INTERVENTION 53
WHEN PEAK CORE TEMP > TCRHOT
LET PEAK CORE TIME = TIME
RESTORE
SILENT
END

INTERVENTION 54
WHEN PEAK CONTMT PRESSURE = PA
LET PEAK PA TIME = TIME
RESTORE
SILENT
END
```

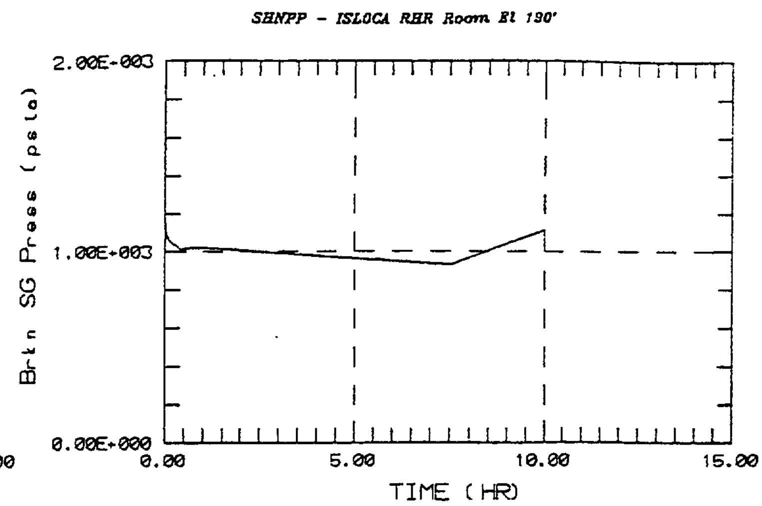
ISLOCA

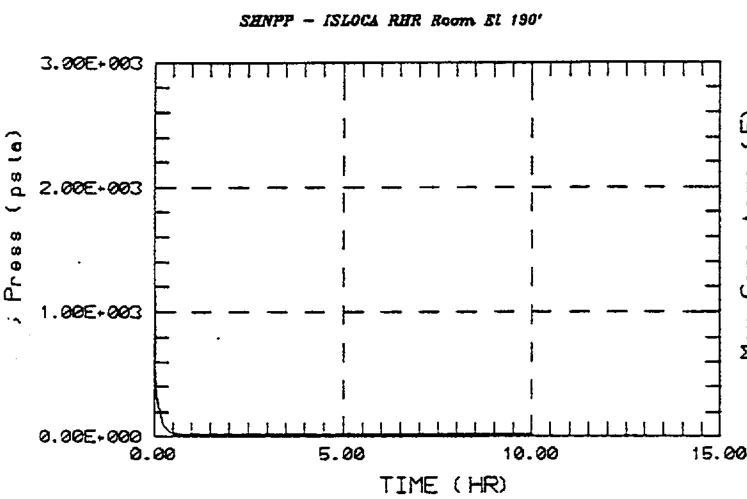
0.0	13	REACTOR SCRAM
0.0	156	MSIV CLOSED
0.0	178	AUX CO2 SUPLY DEPLETD
0.0	190	UHI ACCUM EMPTY
0.0	209	PS BREAK(S) FAILED
0.0	216	HPI FORCED OFF
0.0	217	LPI TRAIN 1 FORCED OFF
0.0	223	PZR SPRAYS FORCED OFF
0.0	224	AUX FEED WATER FORCED OFF
0.0	226	1 PZR HTRS FORCED OFF
0.0	227	MANUAL SCRAM
0.0	228	MAIN FW SHUT OFF
0.0	232	CHARGING PUMPS FORCED OFF
0.0	238	V SEQUENCE
0.1	14	FP MODELS ON
4.7	162	SEC RV OPEN UNBROKEN S/G'S
4.9	15	UNBKN LOOP HOMOGENEOUS
6.3	152	SEC RV OPEN BROKEN S/G
9.1	32	PZR EMPTY
10.8	32	PZR NOT EMPTY
11.8	32	PZR EMPTY
17.2	4	MAIN COOLANT PUMPS OFF
17.2	215	MCP SWITCH OFF OR HI-VIBR TRIP
18.5	15	UNBKN LOOP PHASES SEPARATED
20.4	152	SEC RV NOT OPEN BROKEN S/G
20.4	162	SEC RV NOT OPEN UNBROKEN S/G'S
21.8	152	SEC RV OPEN BROKEN S/G
21.8	162	SEC RV OPEN UNBROKEN S/G'S
23.2	152	SEC RV NOT OPEN BROKEN S/G
23.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
24.7	152	SEC RV OPEN BROKEN S/G
24.7	162	SEC RV OPEN UNBROKEN S/G'S
26.2	152	SEC RV NOT OPEN BROKEN S/G
26.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
27.7	152	SEC RV OPEN BROKEN S/G
27.7	162	SEC RV OPEN UNBROKEN S/G'S
29.2	152	SEC RV NOT OPEN BROKEN S/G
29.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
71.1	32	PZR NOT EMPTY
72.2	32	PZR EMPTY
108.3	32	PZR NOT EMPTY
111.6	32	PZR EMPTY
187.1	32	PZR NOT EMPTY
190.8	32	PZR EMPTY
202.0	32	PZR NOT EMPTY
206.1	32	PZR EMPTY
217.1	32	PZR NOT EMPTY
225.1	32	PZR EMPTY
239.4	32	PZR NOT EMPTY
244.0	32	PZR EMPTY
256.5	32	PZR NOT EMPTY
260.0	32	PZR EMPTY
269.9	32	PZR NOT EMPTY
281.5	32	PZR EMPTY
295.4	32	PZR NOT EMPTY

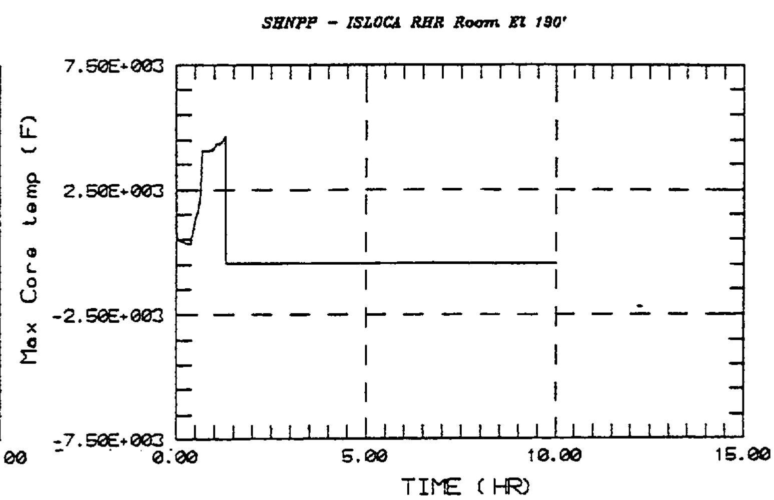
301.9	32	PZR EMPTY
314.6	32	PZR NOT EMPTY
323.0	32	PZR EMPTY
333.2	32	PZR NOT EMPTY
347.4	32	PZR EMPTY
356.1	32	PZR NOT EMPTY
369.4	32	PZR EMPTY
379.9	32	PZR NOT EMPTY
391.4	32	PZR EMPTY
399.5	32	PZR NOT EMPTY
414.3	32	PZR EMPTY
420.2	32	PZR NOT EMPTY
436.6	32	PZR EMPTY
444.2	32	PZR NOT EMPTY
456.2	32	PZR EMPTY
466.1	32	PZR NOT EMPTY
475.1	32	PZR EMPTY
482.2	32	PZR NOT EMPTY
495.0	32	PZR EMPTY
502.4	32	PZR NOT EMPTY
517.9	32	PZR EMPTY
525.0	32	PZR NOT EMPTY
536.8	32	PZR EMPTY
543.8	32	PZR NOT EMPTY
559.0	32	PZR EMPTY
565.6	32	PZR NOT EMPTY
581.2	32	PZR EMPTY
583.1	188	ACCUMULATOR WATER DEPLETED
589.1	32	PZR NOT EMPTY
598.5	32	PZR EMPTY
1273.6	25	PS NONEQ THERMO
1297.5	49	CORE HAS UNCOV
2397.2	176	BURN IN AUX BLDG
2405.1	176	NO BURN IN AUX BLDG
2423.5	176	BURN IN AUX BLDG
2480.6	176	NO BURN IN AUX BLDG
2480.7	176	BURN IN AUX BLDG
2567.1	176	NO BURN IN AUX BLDG
2668.8	176	BURN IN AUX BLDG
2676.8	176	NO BURN IN AUX BLDG
2773.4	176	BURN IN AUX BLDG
2781.8	176	NO BURN IN AUX BLDG
2877.3	176	BURN IN AUX BLDG
2885.2	176	NO BURN IN AUX BLDG
3025.1	176	BURN IN AUX BLDG
3032.9	176	NO BURN IN AUX BLDG
3278.1	176	BURN IN AUX BLDG
3285.6	176	NO BURN IN AUX BLDG
3316.0	176	BURN IN AUX BLDG
3328.6	176	NO BURN IN AUX BLDG
3352.9	176	BURN IN AUX BLDG
3362.1	176	NO BURN IN AUX BLDG
3388.8	176	BURN IN AUX BLDG
3398.4	176	NO BURN IN AUX BLDG
4565.0	2	SUPPORT PLATE FAILED
4625.0	3	RV FAILED
4625.2	61	CORIUM IN CAVITY

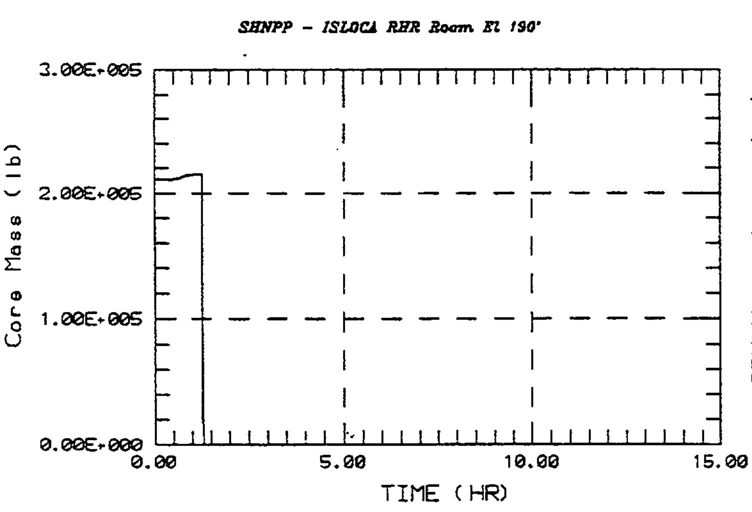
4663.2	57	WATER IN CAVITY
4671.8	28	DWNCMR NOT BLCKD FOR GAS XPORT
4706.4	81	WATER ON LOWER CMPT FLOOR
5360.5	79	FANS/COOLERS ON
5654.8	57	CAVITY DRY
29177.7	75	BURN IN PROGRESS IN LOWER CMPT
29185.0	75	NO BURN IN LOWER CMPT
31264.7	75	BURN IN PROGRESS IN LOWER CMPT
31271.7	75	NO BURN IN LOWER CMPT
31636.0	75	BURN IN PROGRESS IN LOWER CMPT
31642.6	75	NO BURN IN LOWER CMPT
34703.9	75	BURN IN PROGRESS IN LOWER CMPT
34719.4	75	NO BURN IN LOWER CMPT
34986.8	75	BURN IN PROGRESS IN LOWER CMPT
35002.7	75	NO BURN IN LOWER CMPT

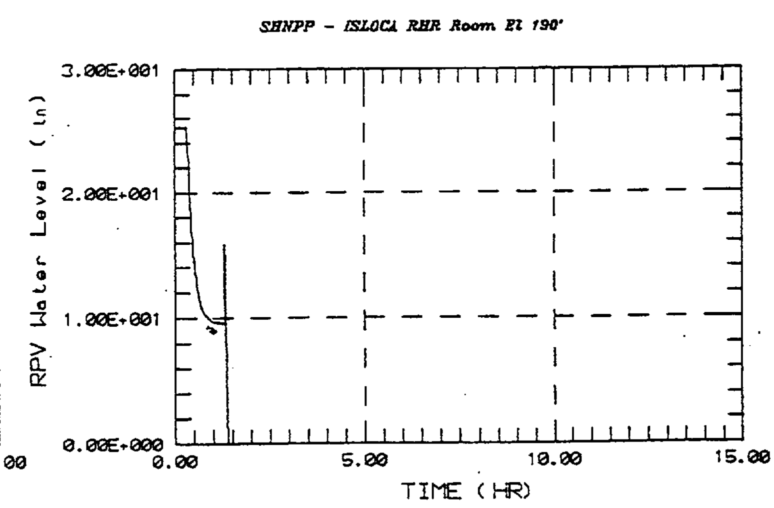

SHNPP - ISLOCA RHR Room El 190'
Core H2 Mass (
7.50E+002
5.00E+002
2.50E+002
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

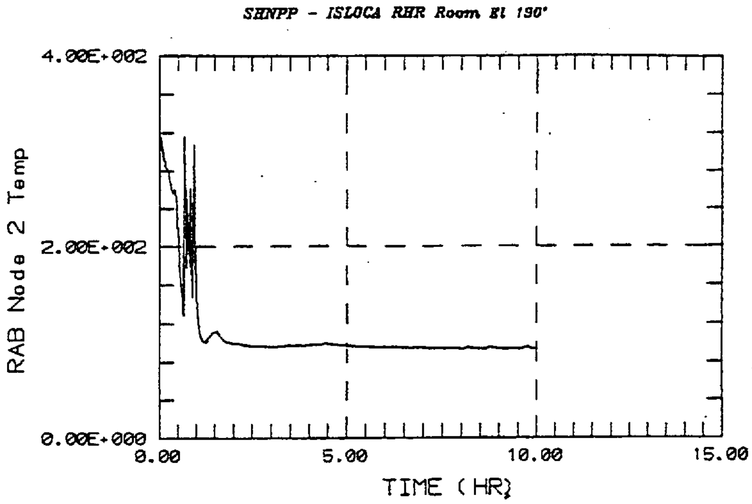

SHNPP - ISLOCA RHR Room El 190'
Brkn SG Press (psia)
2.00E+003
1.00E+003
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

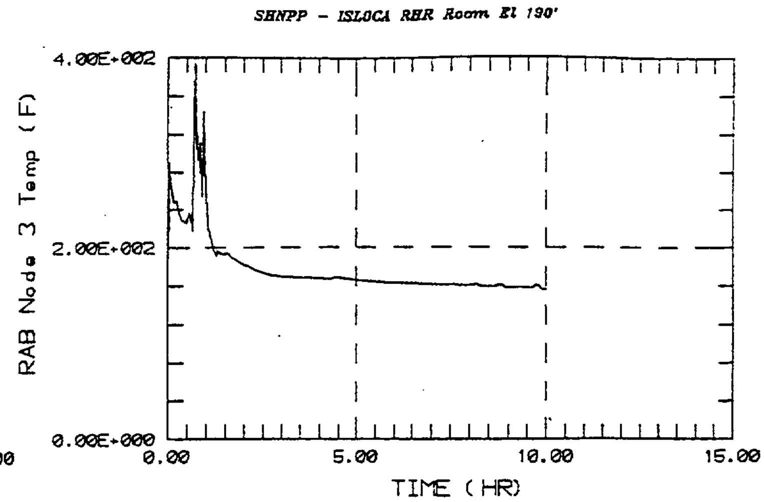

SHNPP - ISLOCA RHR Room El 190'
Press (psia)
3.00E+003
2.00E+003
1.00E+003
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

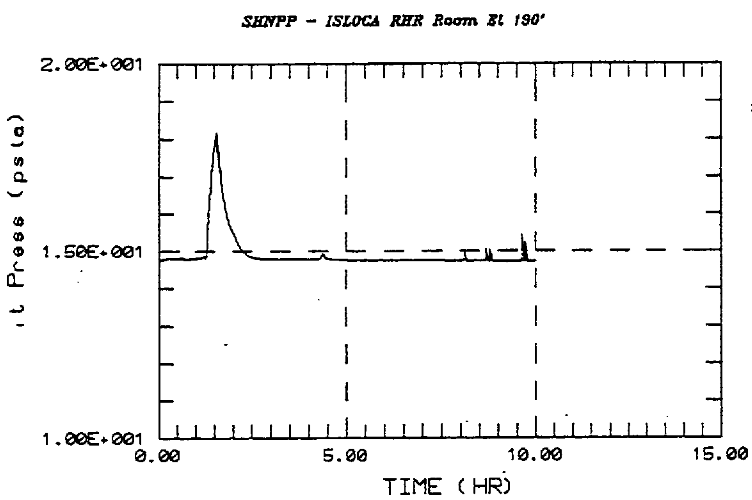

SHNPP - ISLOCA RHR Room El 190'
Max Core Temp (F)
7.50E+003
2.50E+003
-2.50E+003
-7.50E+003
0.00
5.00
10.00
15.00
TIME (HR)

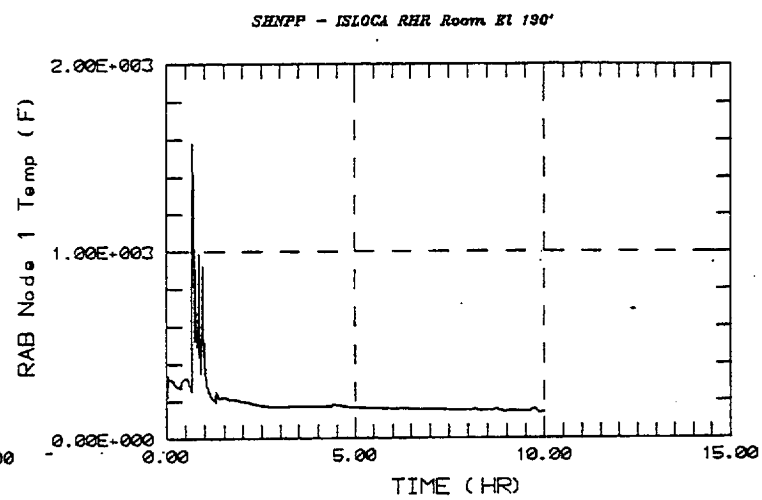

SHNPP - ISLOCA RHR Room El 190'
Core Mass (lb)
3.00E+005
2.00E+005
1.00E+005
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

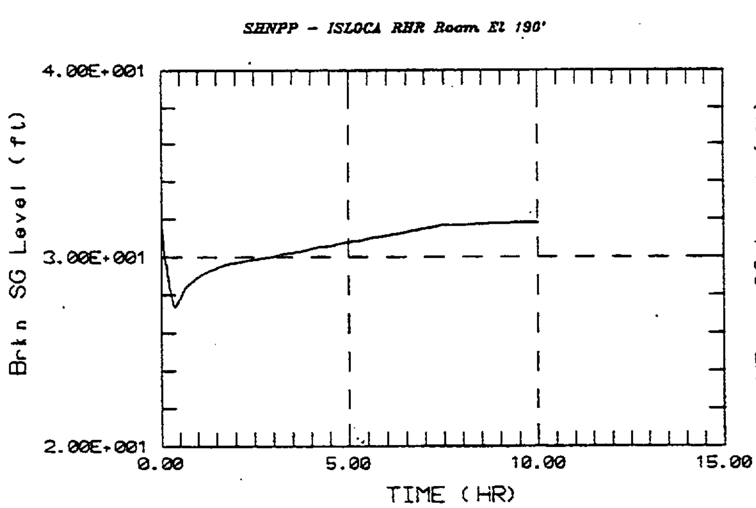

SHNPP - ISLOCA RHR Room El 190'
RPV Water Level (in)
3.00E+001
2.00E+001
1.00E+001
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

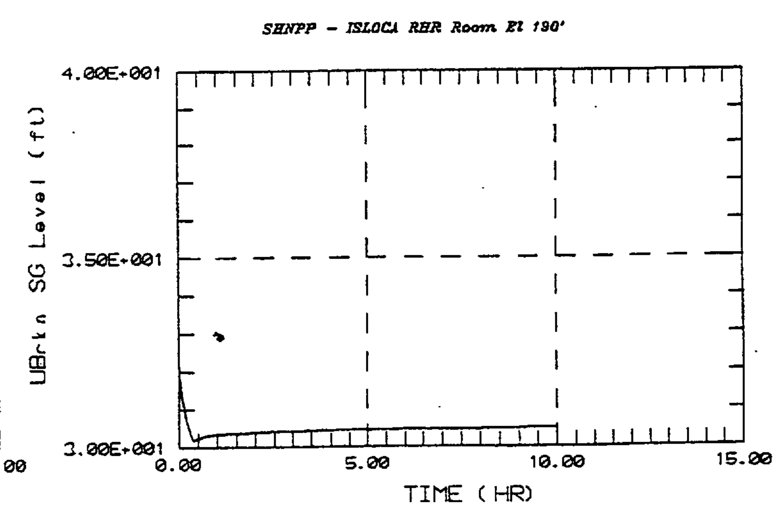

SHNPP - ISLOCA RHR Room El 190'
RAB Node 2 Temp
4.00E+002
2.00E+002
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

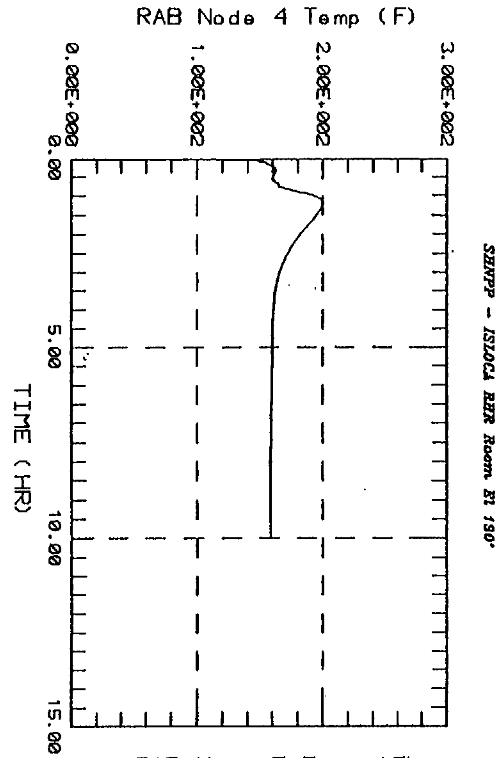

SHNPP - ISLOCA RHR Room El 190'
RAB Node 3 Temp (F)
4.00E+002
2.00E+002
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

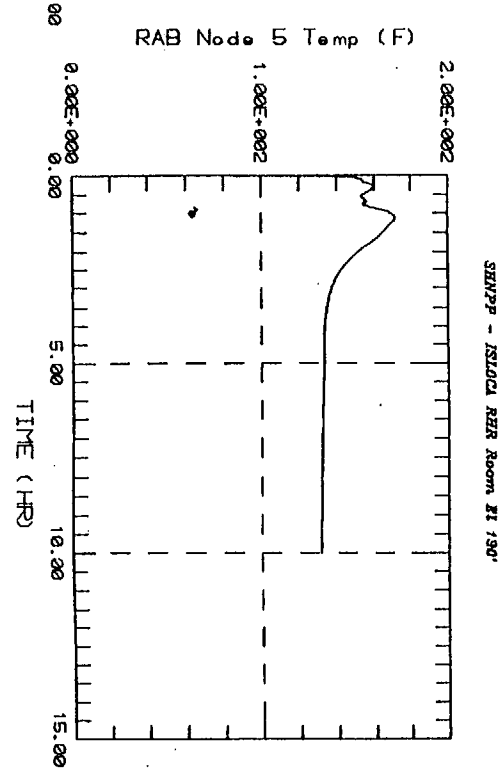

SHNPP - ISLOCA RHR Room El 190'
ıt Press (psia)
2.00E+001
1.50E+001
1.00E+001
0.00
5.00
10.00
15.00
TIME (HR)

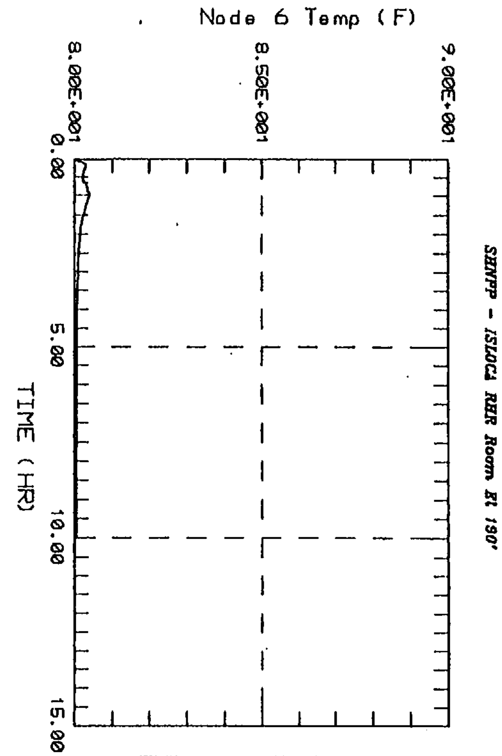

SHNPP - ISLOCA RHR Room El 190'
RAB Node 1 Temp (F)
2.00E+003
1.00E+003
0.00E+000
0.00
5.00
10.00
15.00
TIME (HR)

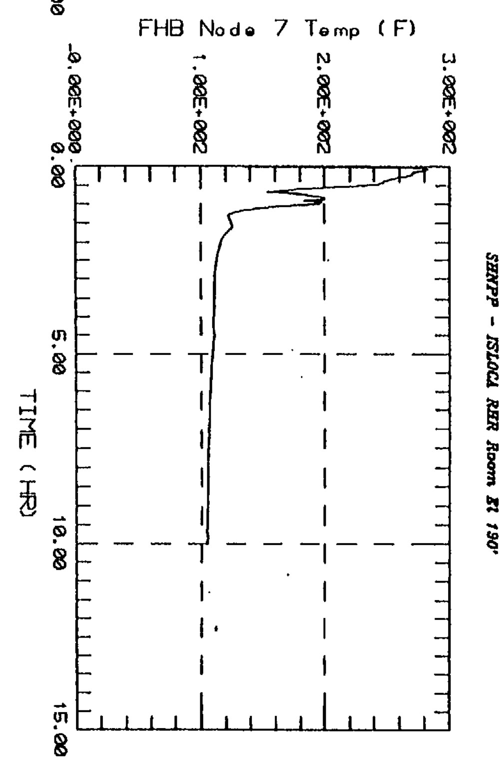

SHNPP - ISLOCA RHR Room El 190'
Brkn SG Level (ft)
4.00E+001
3.00E+001
2.00E+001
0.00
5.00
10.00
15.00
TIME (HR)

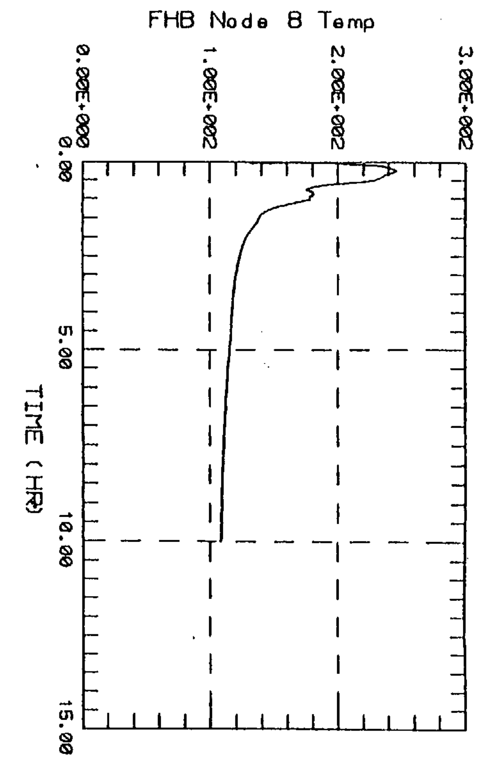

SHNPP - ISLOCA RHR Room El 190'
UBrkn SG Level (ft)
4.00E+001
3.50E+001
3.00E+001
0.00
5.00
10.00
15.00
TIME (HR)

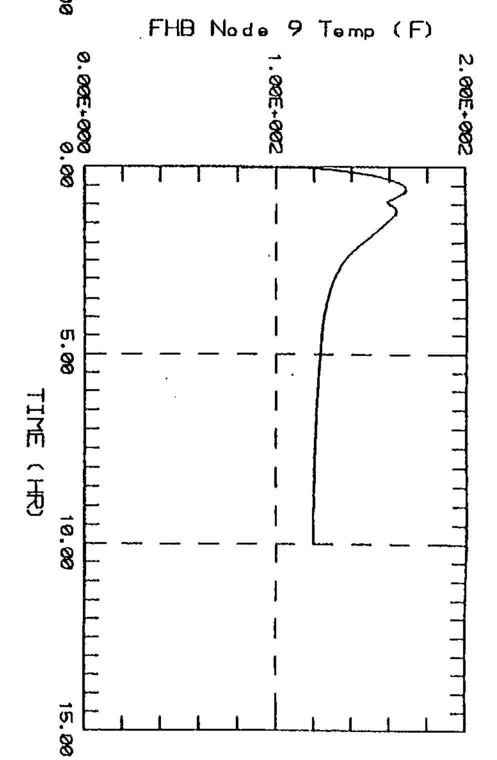

RAB Node 4 Temp (F)
0.00E+000
1.00E+002
2.00E+002
3.00E+002
0.00
5.00
10.00
15.00
TIME (HR)
SHNPP - ISLOCA RHR Room El 190'


RAB Node 5 Temp (F)
0.00E+000
1.00E+002
2.00E+002
0.00
5.00
10.00
15.00
TIME (HR)
SHNPP - ISLOCA RHR Room El 190'

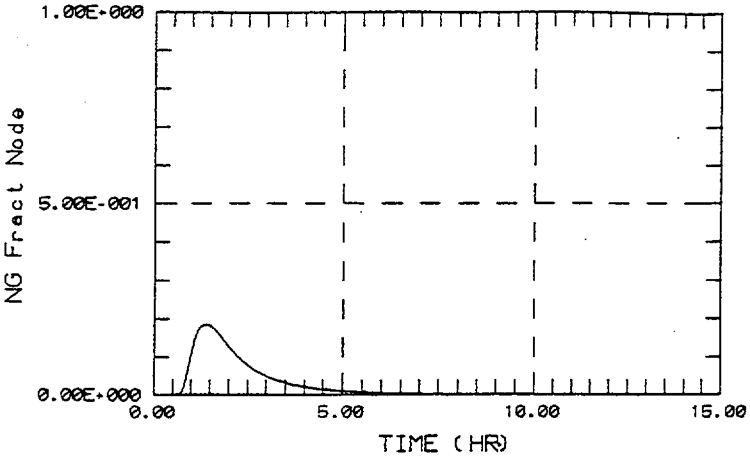

Node 6 Temp (F)
8.00E+001
8.50E+001
9.00E+001
0.00
5.00
10.00
15.00
TIME (HR)
SHNPP - ISLOCA RHR Room El 190'


FHB Node 7 Temp (F)
-0.00E+000
1.00E+002
2.00E+002
3.00E+002
0.00
5.00
10.00
15.00
TIME (HR)
SHNPP - ISLOCA RHR Room El 190'

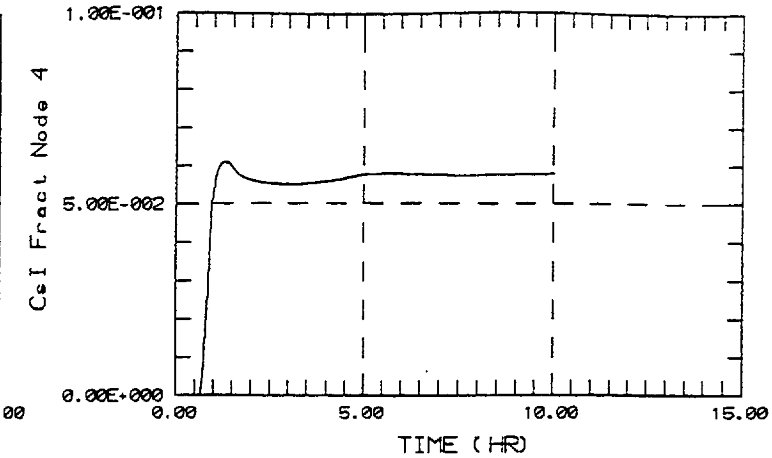

FHB Node 8 Temp
0.00E+000
1.00E+002
2.00E+002
3.00E+002
0.00
5.00
10.00
15.00
TIME (HR)
SHNPP - ISLOCA RHR Room El 190'


FHB Node 9 Temp (F)
0.00E+000
1.00E+002
2.00E+002
0.00
5.00
10.00
15.00
TIME (HR)
SHNPP - ISLOCA RHR Room El 190'

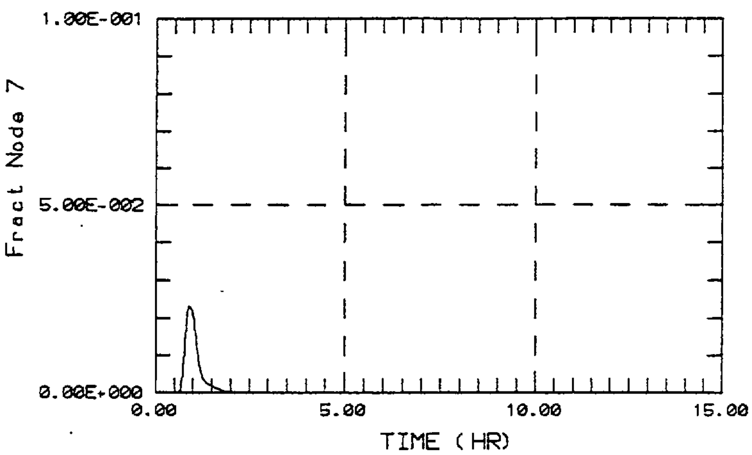
SHNPP - ISLOCA RHR Room El 190'



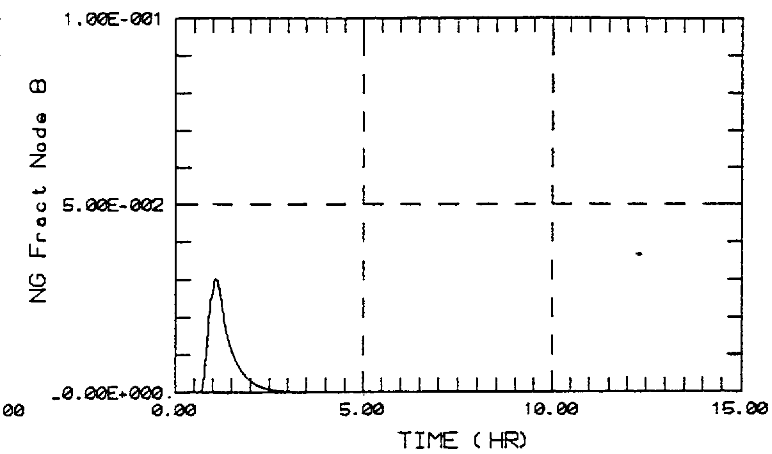
SHNPP - ISLOCA RHR Room El 190'



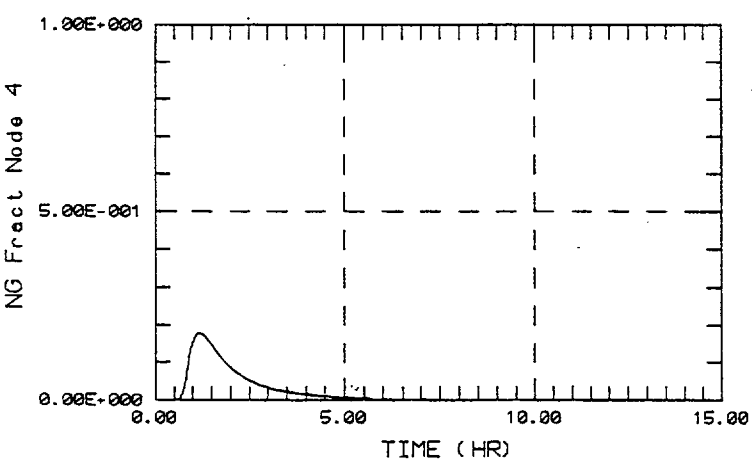
SHNPP - ISLOCA RHR Room El 190'



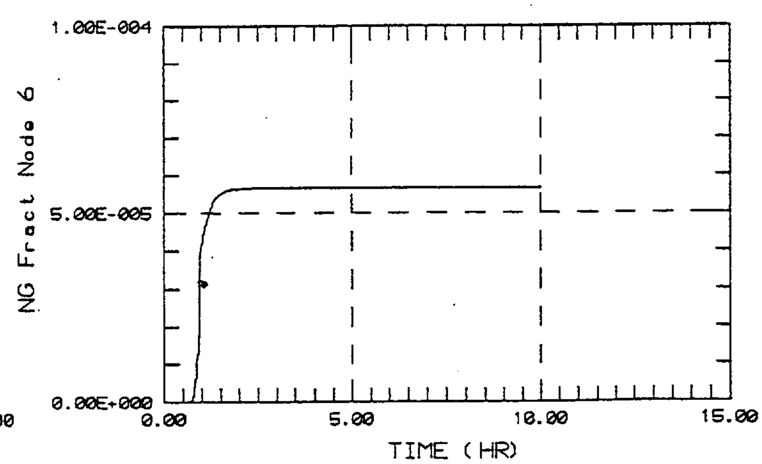
SHNPP - ISLOCA RHR Room El 190'



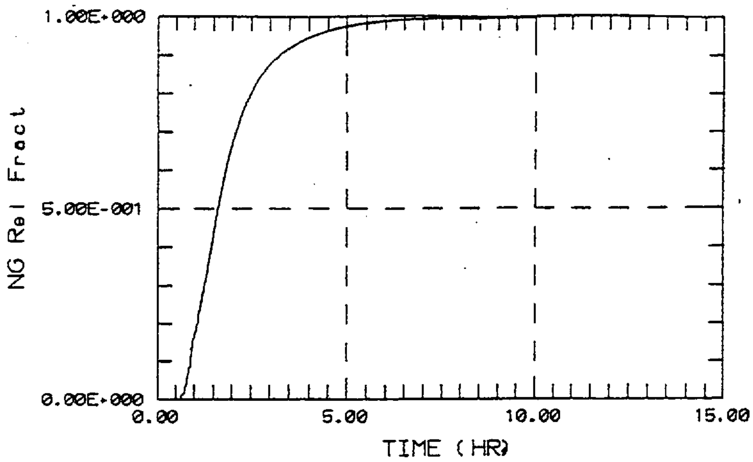
SHNPP - ISLOCA RHR Room El 190'



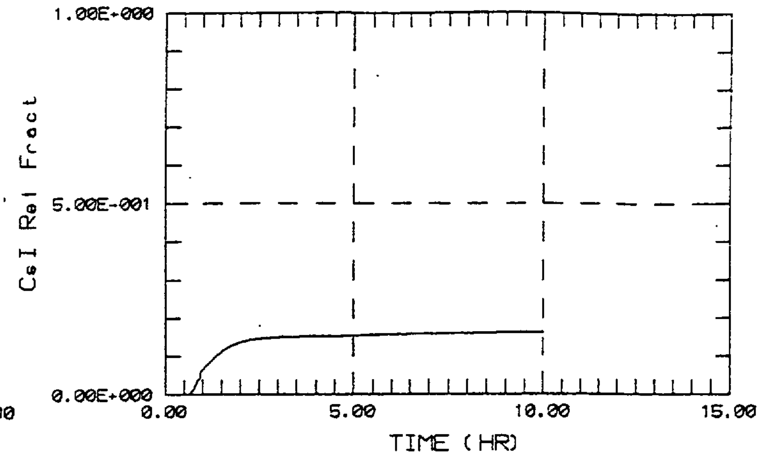
SHNPP - ISLOCA RHR Room El 190'



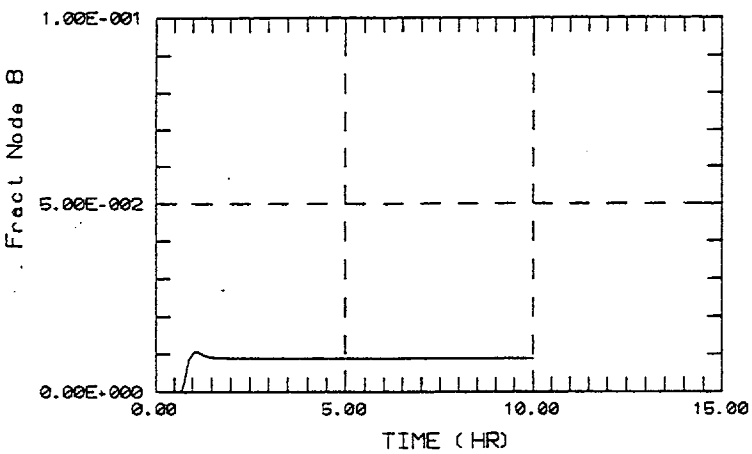
SHNPP – ISLOCA RHR Room El 190'



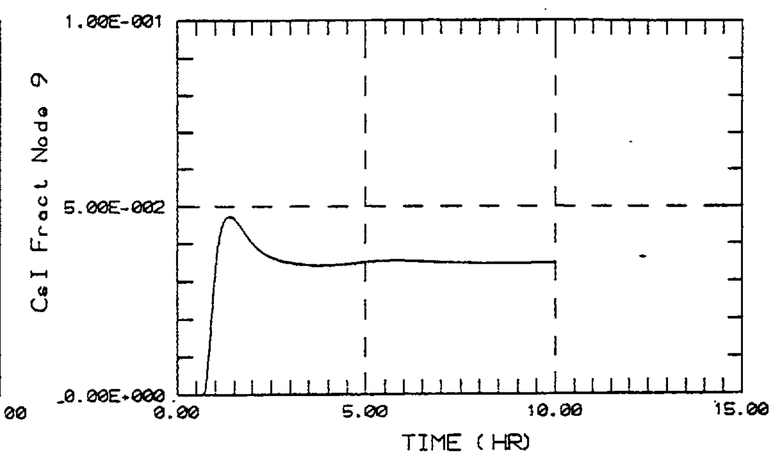
SHNPP – ISLOCA RHR Room El 190'



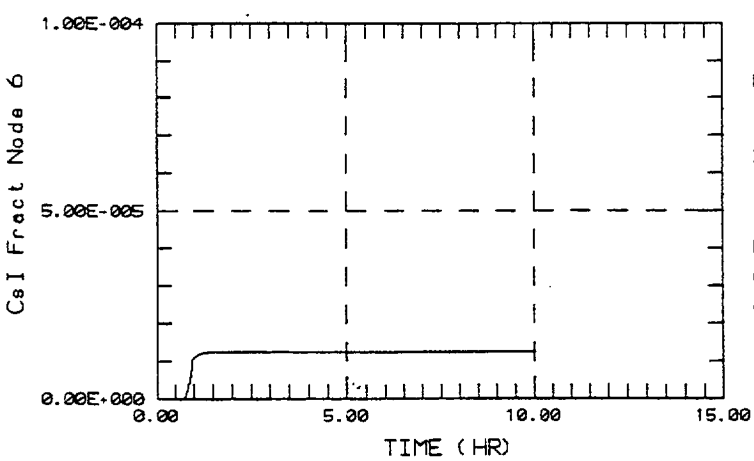
SHNPP – ISLOCA RHR Room El 190'



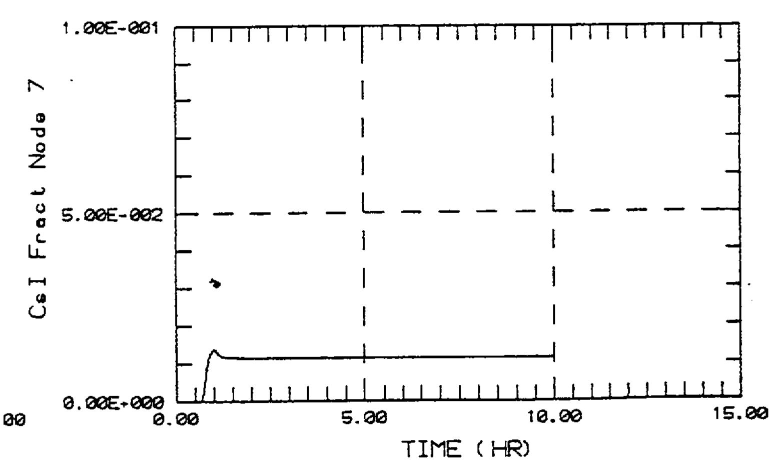
SHNPP – ISLOCA RHR Room El 190'



SHNPP – ISLOCA RHR Room El 190'



SHNPP – ISLOCA RHR Room El 190'



```
C TEST
TITLE
        SHNPP - SGTR
        END TITLE

ATTACH ATTACH.SAM

PARAMETER CHANGES
        SGTR BREAK NODE 4
        SGTR BREAK ELEV 3.3 FT
        SGTR BREAK AREA 5.454E-3 FT**2
        BREAK AREA 0.0 M**2
        TDMAX    5  SECONDS
        END OF PARAMETER CHANGES AND NOLIST

NOT A RESTART

PRINT TIME 5 HOURS

FINAL TIME 20.0

PARALLEL

WHEN BEGIN
     SCRAM ON
     BREAK ON
     HPI OFF
     MAIN FED WATER PUMPS OFF
     PZR HEATERS OFF
     PZR SPRAYS OFF
     END

WHEN TIME > 1 MIN
PARAMETER CHANGE
  WAFWXB 0.
  FARVBX 0.
END
END

WHEN PBS > $PSGSVL$
     LABEL: SG SRV STUCK OPEN
     IEVNT(239) ON
     END

WHEN ZWRWST < 9.31 FT
     LPI ON
     RECIRCULATION MODE ON
     END

INTERVENTION 47
WHEN SCRAM IS TRUE
 PARAMETER CHANGE
  ZWCTLB 41. FT
  ZWCTLU 41. FT
  END
END
```

```
INTERVENTION 49
WHEN RPV FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 50
WHEN CONTMT FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 51
WHEN CORE UNCOVERED IS TRUE
LET CORE UNC TIME = TIME
END

INTERVENTION 52
WHEN HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
  OR HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
LET MELT ONSET TIME = TIME
END

INTERVENTION 53
WHEN PEAK CORE TEMP > TCRHOT
LET PEAK CORE TIME = TIME
RESTORE
SILENT
END

INTERVENTION 54
WHEN PEAK CONTMT PRESSURE = PA
LET PEAK PA TIME = TIME
RESTORE
SILENT
END

INTERVENTION 55
WHEN IEVNT(103) IS TRUE
 OR IEVNT(79) IS TRUE
 PARAMETER CHANGE
  FLPHI  10.0
END
RESTORE 56
END

INTERVENTION 56
WHEN IEVNT(103) IS FALSE
 AND IEVNT(79) IS FALSE
 PARAMETER CHANGE
  FLPHI  2.0
END
RESTORE 55
END

INTERVENTION 57
```

```
WHEN IEVNT(42) IS TRUE
 INCREMENT PZR SVS OPEN BY 1
RESTORE 58
END

INTERVENTION 58
WHEN IEVNT(42) IS FALSE
RESTORE 57
END

WHEN PA > 99.7 PSI
LET CONTMT OVERPRESS TIME = TIME
FULL OUTPUT
REPORT
END
```

SGTR

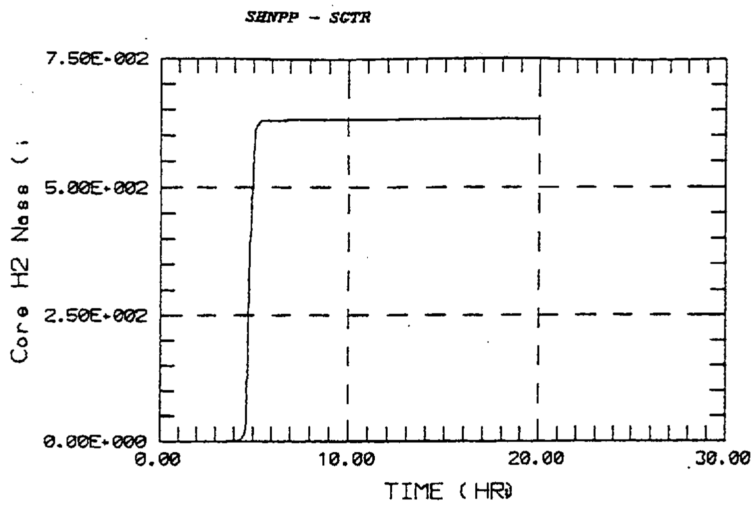
0.0	13	REACTOR SCRAM
0.0	154	AUX FEEDWATER ON
0.0	156	MSIV CLOSED
0.0	178	AUX CO2 SUPLY DEPLETD
0.0	190	UHI ACCUM EMPTY
0.0	209	PS BREAK(S) FAILED
0.0	216	HPI FORCED OFF
0.0	223	PZR SPRAYS FORCED OFF
0.0	226	1 PZR HTRS FORCED OFF
0.0	227	MANUAL SCRAM
0.0	228	MAIN FW SHUT OFF
5.1	162	SEC RV OPEN UNBROKEN S/G'S
6.0	152	SEC RV OPEN BROKEN S/G
53.1	162	SEC RV NOT OPEN UNBROKEN S/G'S
54.9	162	SEC RV OPEN UNBROKEN S/G'S
56.9	152	SEC RV NOT OPEN BROKEN S/G
56.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
58.5	152	SEC RV OPEN BROKEN S/G
58.5	162	SEC RV OPEN UNBROKEN S/G'S
61.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
63.8	162	SEC RV OPEN UNBROKEN S/G'S
65.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
67.3	162	SEC RV OPEN UNBROKEN S/G'S
69.1	162	SEC RV NOT OPEN UNBROKEN S/G'S
86.8	162	(20) SEC RV NOT OPEN UNBROKEN S/G'S
105.0	162	(30) SEC RV NOT OPEN UNBROKEN S/G'S
106.8	6	LPI ON
122.6	162	(40) SEC RV NOT OPEN UNBROKEN S/G'S
126.2	153	SEC SV(S) OPEN BROKEN S/G
126.2	239	MODEL DEVELPMNT USE
133.7	152	SEC RV NOT OPEN BROKEN S/G
140.6	162	(50) SEC RV NOT OPEN UNBROKEN S/G'S
230.5	32	PZR EMPTY
240.9	32	PZR NOT EMPTY
245.9	32	PZR EMPTY
271.5	32	PZR NOT EMPTY
274.3	32	PZR EMPTY
283.2	15	UNBKN LOOP HOMOGENEOUS
287.5	32	PZR NOT EMPTY
297.5	32	PZR EMPTY
855.4	162	(60) SEC RV NOT OPEN UNBROKEN S/G'S
1759.4	162	(70) SEC RV NOT OPEN UNBROKEN S/G'S
1984.9	4	MAIN COOLANT PUMPS OFF
1984.9	215	MCP SWITCH OFF OR HI-VIBR TRIP
1989.9	15	UNBKN LOOP PHASES SEPARATED
2015.6	32	PZR NOT EMPTY
2333.5	32	PZR EMPTY
2585.4	162	(80) SEC RV NOT OPEN UNBROKEN S/G'S
3369.9	32	PZR NOT EMPTY
7574.9	151	BROKEN S/G DRY
8398.8	162	SEC RV OPEN UNBROKEN S/G'S
8400.0	162	SEC RV NOT OPEN UNBROKEN S/G'S
9293.4	162	SEC RV OPEN UNBROKEN S/G'S
9294.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
9295.7	162	SEC RV OPEN UNBROKEN S/G'S

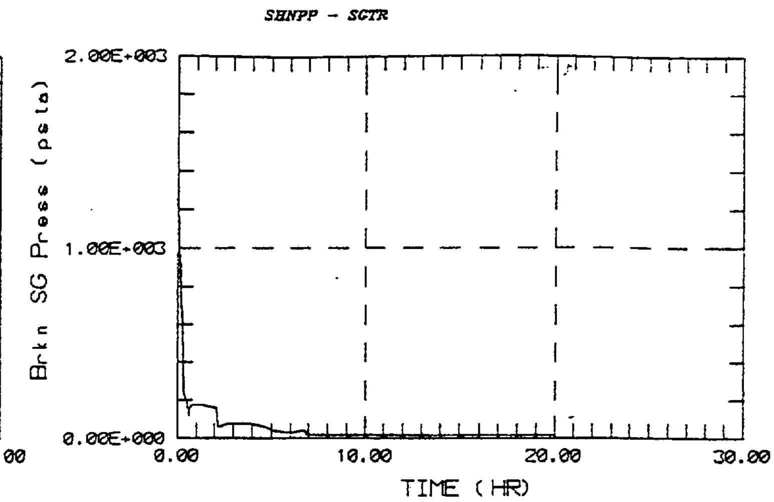
9296.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
9298.4	162	SEC RV OPEN UNBROKEN S/G'S
9299.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
9306.6	162	SEC RV OPEN UNBROKEN S/G'S
9307.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
9404.7	162	(20) SEC RV NOT OPEN UNBROKEN S/G'S
9550.8	162	(30) SEC RV NOT OPEN UNBROKEN S/G'S
9609.2	162	(40) SEC RV NOT OPEN UNBROKEN S/G'S
9687.3	162	(50) SEC RV NOT OPEN UNBROKEN S/G'S
9857.4	162	(60) SEC RV NOT OPEN UNBROKEN S/G'S
9924.7	162	(70) SEC RV NOT OPEN UNBROKEN S/G'S
10042.8	162	(80) SEC RV NOT OPEN UNBROKEN S/G'S
10161.4	162	(90) SEC RV NOT OPEN UNBROKEN S/G'S
10224.1	162	(100) SEC RV NOT OPEN UNBROKEN S/G'S
10751.9	162	(150) SEC RV NOT OPEN UNBROKEN S/G'S
11246.3	162	(200) SEC RV NOT OPEN UNBROKEN S/G'S
11971.5	162	(250) SEC RV NOT OPEN UNBROKEN S/G'S
12698.1	25	PS NONEQ THERMO
12828.3	162	(300) SEC RV NOT OPEN UNBROKEN S/G'S
13069.0	14	FP MODELS ON
13069.0	49	CORE HAS UNCOV
13326.9	32	PZR EMPTY
13658.1	162	(350) SEC RV NOT OPEN UNBROKEN S/G'S
15349.3	162	(400) SEC RV NOT OPEN UNBROKEN S/G'S
19645.0	162	SEC RV OPEN UNBROKEN S/G'S
19646.3	162	SEC RV NOT OPEN UNBROKEN S/G'S
19647.6	162	SEC RV OPEN UNBROKEN S/G'S
19648.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
24487.5	2	SUPPORT PLATE FAILED
24547.5	3	RV FAILED
24547.6	61	CORIUM IN CAVITY
24551.2	59	WATER FLOODING IN CAVITY TO B
24551.2	59	WATER NOT FLOODING IN CAVITY TO B
24551.2	27	UNBKN LOOPS NOT BLOCKED AT PUMP BOWLS
24551.2	59	WATER FLOODING IN CAVITY TO B
24551.2	59	WATER NOT FLOODING IN CAVITY TO B
24551.2	59	WATER FLOODING IN CAVITY TO B
24551.2	58	CORIUM FLOODING IN CAVITY TO B
24551.2	82	CORIUM IN LOWER CMPT
24551.2	57	WATER IN CAVITY
24551.4	81	WATER ON LOWER CMPT FLOOR
24551.4	79	FANS/COOLERS ON
24551.4	75	BURN IN PROGRESS IN LOWER CMPT
24553.1	28	DWNCMR NOT BLCKD FOR GAS XPORT
24554.5	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.5	59	WATER NOT FLOODING IN CAVITY TO B
24554.5	58	CORIUM FLOODING IN CAVITY TO B
24554.5	59	WATER FLOODING IN CAVITY TO B
24554.5	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.6	58	CORIUM FLOODING IN CAVITY TO B
24554.6	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.6	59	WATER NOT FLOODING IN CAVITY TO B
24554.6	58	CORIUM FLOODING IN CAVITY TO B
24554.6	59	WATER FLOODING IN CAVITY TO B
24554.6	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.6	59	WATER NOT FLOODING IN CAVITY TO B
24554.6	58	CORIUM FLOODING IN CAVITY TO B

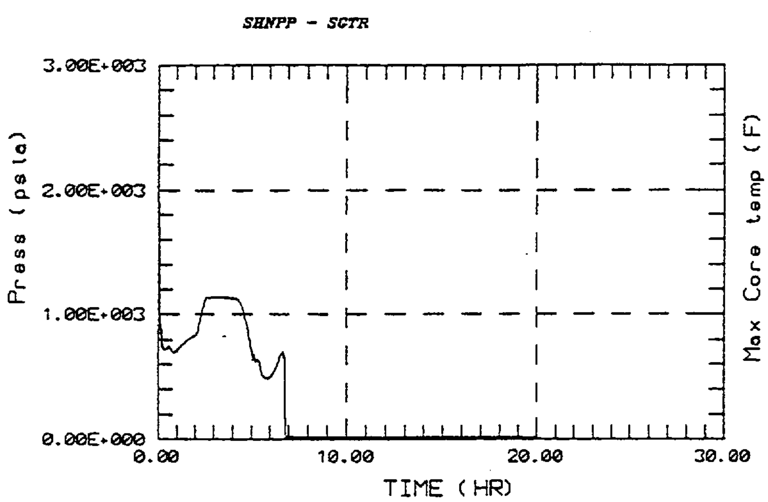
24554.6	59	WATER FLOODING IN CAVITY TO B
24554.6	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.7	58	CORIUM FLOODING IN CAVITY TO B
24554.7	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.8	58	CORIUM FLOODING IN CAVITY TO B
24554.8	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.8	59	WATER NOT FLOODING IN CAVITY TO B
24554.8	58	CORIUM FLOODING IN CAVITY TO B
24554.8	59	WATER FLOODING IN CAVITY TO B
24554.8	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.8	59	WATER NOT FLOODING IN CAVITY TO B
24554.8	58	CORIUM FLOODING IN CAVITY TO B
24554.8	59	WATER FLOODING IN CAVITY TO B
24554.8	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.9	58	CORIUM FLOODING IN CAVITY TO B
24554.9	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.9	59	WATER NOT FLOODING IN CAVITY TO B
24554.9	58	CORIUM FLOODING IN CAVITY TO B
24554.9	59	WATER FLOODING IN CAVITY TO B
24554.9	58	CORIUM NOT FLOODING IN CAVITY TO B
24554.9	59	WATER NOT FLOODING IN CAVITY TO B
24554.9	58	CORIUM FLOODING IN CAVITY TO B
24554.9	59	WATER FLOODING IN CAVITY TO B
24554.9	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.0	58	CORIUM FLOODING IN CAVITY TO B
24555.0	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.0	59	WATER NOT FLOODING IN CAVITY TO B
24555.0	58	CORIUM FLOODING IN CAVITY TO B
24555.0	59	WATER FLOODING IN CAVITY TO B
24555.0	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.0	59	WATER NOT FLOODING IN CAVITY TO B
24555.0	58	CORIUM FLOODING IN CAVITY TO B
24555.0	59	WATER FLOODING IN CAVITY TO B
24555.0	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.0	59	WATER NOT FLOODING IN CAVITY TO B
24555.0	58	CORIUM FLOODING IN CAVITY TO B
24555.0	59	WATER FLOODING IN CAVITY TO B
24555.1	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.1	58	CORIUM FLOODING IN CAVITY TO B
24555.1	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.1	59	WATER NOT FLOODING IN CAVITY TO B
24555.1	58	CORIUM FLOODING IN CAVITY TO B
24555.1	59	WATER FLOODING IN CAVITY TO B
24555.2	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.2	59	WATER NOT FLOODING IN CAVITY TO B
24555.2	58	CORIUM FLOODING IN CAVITY TO B
24555.2	59	WATER FLOODING IN CAVITY TO B
24555.2	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.2	59	WATER NOT FLOODING IN CAVITY TO B
24555.2	58	CORIUM FLOODING IN CAVITY TO B
24555.2	59	WATER FLOODING IN CAVITY TO B
24555.2	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.2	59	WATER NOT FLOODING IN CAVITY TO B
24555.2	58	CORIUM FLOODING IN CAVITY TO B
24555.2	59	WATER FLOODING IN CAVITY TO B
24555.2	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.2	59	WATER NOT FLOODING IN CAVITY TO B

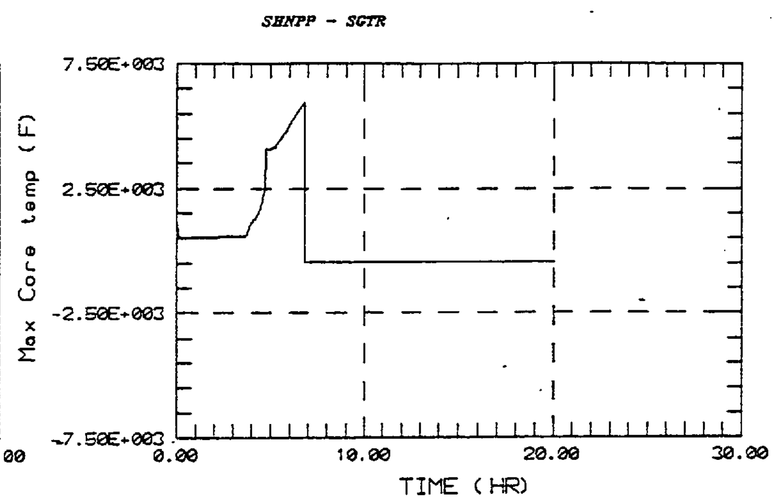
24555.2	58	CORIUM FLOODING IN CAVITY TO B
24555.2	59	WATER FLOODING IN CAVITY TO B
24555.2	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.2	59	WATER NOT FLOODING IN CAVITY TO B
24555.3	58	CORIUM FLOODING IN CAVITY TO B
24555.3	59	WATER FLOODING IN CAVITY TO B
24555.3	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.3	59	WATER NOT FLOODING IN CAVITY TO B
24555.3	58	CORIUM FLOODING IN CAVITY TO B
24555.3	59	WATER FLOODING IN CAVITY TO B
24555.3	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.3	59	WATER NOT FLOODING IN CAVITY TO B
24555.3	58	CORIUM FLOODING IN CAVITY TO B
24555.3	59	WATER FLOODING IN CAVITY TO B
24555.3	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.5	58	CORIUM FLOODING IN CAVITY TO B
24555.5	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.6	58	CORIUM FLOODING IN CAVITY TO B
24555.6	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.7	58	CORIUM FLOODING IN CAVITY TO B
24555.7	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.8	58	CORIUM FLOODING IN CAVITY TO B
24555.8	58	CORIUM NOT FLOODING IN CAVITY TO B
24555.9	58	CORIUM FLOODING IN CAVITY TO B
24555.9	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.0	58	CORIUM FLOODING IN CAVITY TO B
24556.0	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.1	58	CORIUM FLOODING IN CAVITY TO B
24556.1	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.2	58	CORIUM FLOODING IN CAVITY TO B
24556.2	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.3	58	CORIUM FLOODING IN CAVITY TO B
24556.3	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.4	58	CORIUM FLOODING IN CAVITY TO B
24556.5	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.5	75	NO BURN IN LOWER CMPT
24556.6	58	CORIUM FLOODING IN CAVITY TO B
24556.6	58	CORIUM NOT FLOODING IN CAVITY TO B
24556.7	58	CORIUM FLOODING IN CAVITY TO B
24556.9	58	CORIUM NOT FLOODING IN CAVITY TO B
24560.3	59	WATER NOT FLOODING IN CAVITY TO B
24560.3	59	WATER FLOODING IN CAVITY TO B
24560.4	59	WATER NOT FLOODING IN CAVITY TO B
24560.4	59	WATER FLOODING IN CAVITY TO B
24560.4	59	WATER NOT FLOODING IN CAVITY TO B
24576.6	188	ACCUMULATOR WATER DEPLETED
24589.6	27	UNBKN LOOPS BLOCKED
24813.9	103	CONTMT SPRAYS ON
27470.0	181	RECIRC SYSTEM IN OPERATION
27470.0	213	LPI SWITCH TRAIN 1: MAN ON
27470.0	220	RECIRC SWITCH: MAN ON
29066.0	162	SEC RV OPEN UNBROKEN S/G'S
29067.3	162	SEC RV NOT OPEN UNBROKEN S/G'S
29068.6	162	SEC RV OPEN UNBROKEN S/G'S
29069.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
35515.9	65	CAV CPLD MODEL USED
45081.8	162	SEC RV OPEN UNBROKEN S/G'S

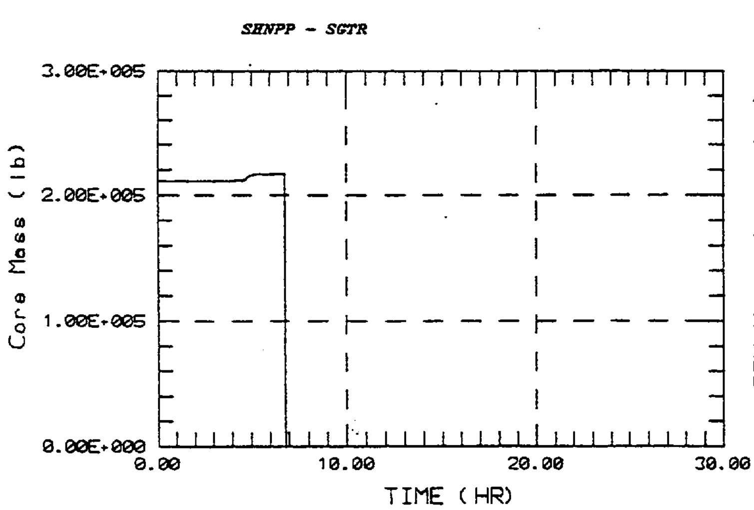
45082.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
45084.6	162	SEC RV OPEN UNBROKEN S/G'S
45085.9	162	SEC RV NOT OPEN UNBROKEN S/G'S

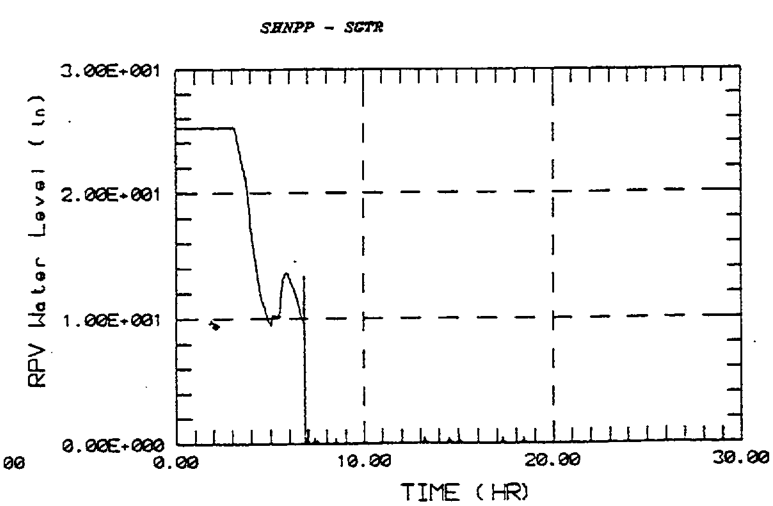

SHNPP - SGTR
Core H2 Mass (i
7.50E+002
5.00E+002
2.50E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

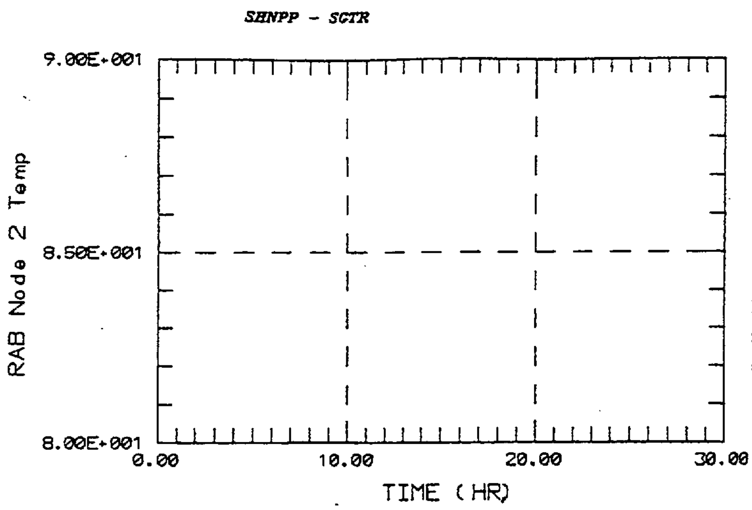

SHNPP - SGTR
Brkn SG Press (psia)
2.00E+003
1.00E+003
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

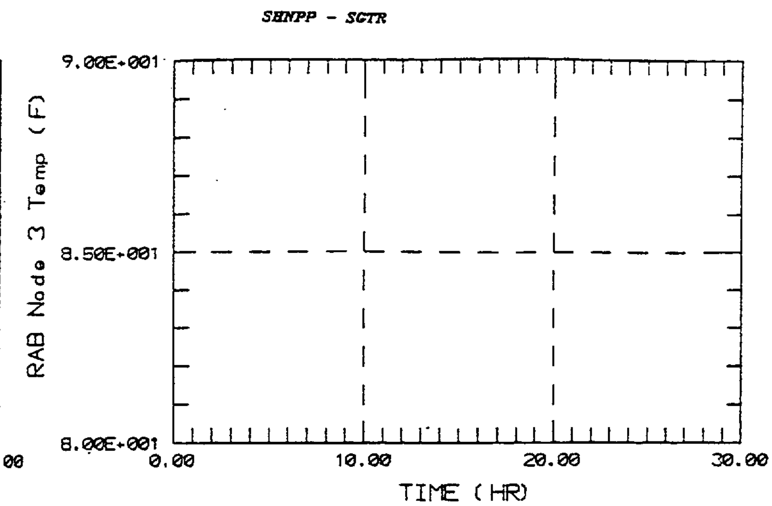

SHNPP - SGTR
Press (psia)
3.00E+003
2.00E+003
1.00E+003
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

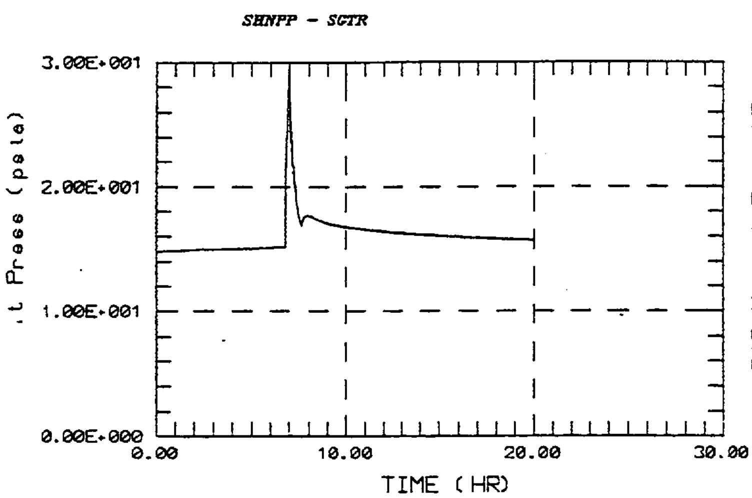

SHNPP - SGTR
Max Core temp (F)
7.50E+003
2.50E+003
-2.50E+003
-7.50E+003
0.00
10.00
20.00
30.00
TIME (HR)

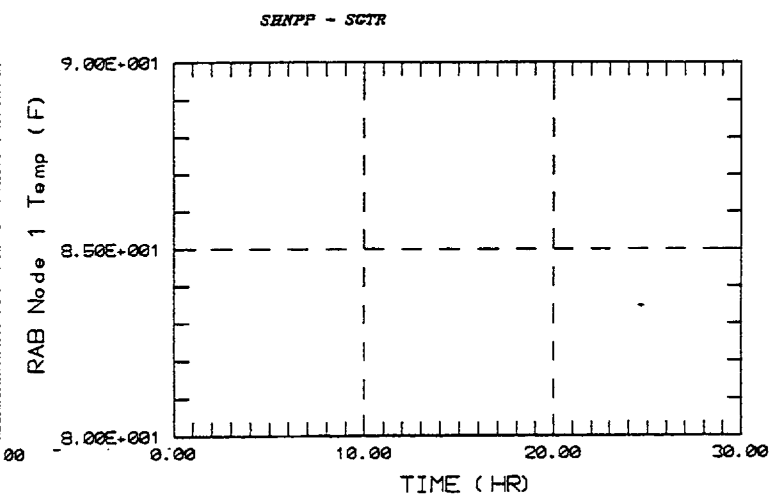

SHNPP - SGTR
Core Mass (lb)
3.00E+005
2.00E+005
1.00E+005
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

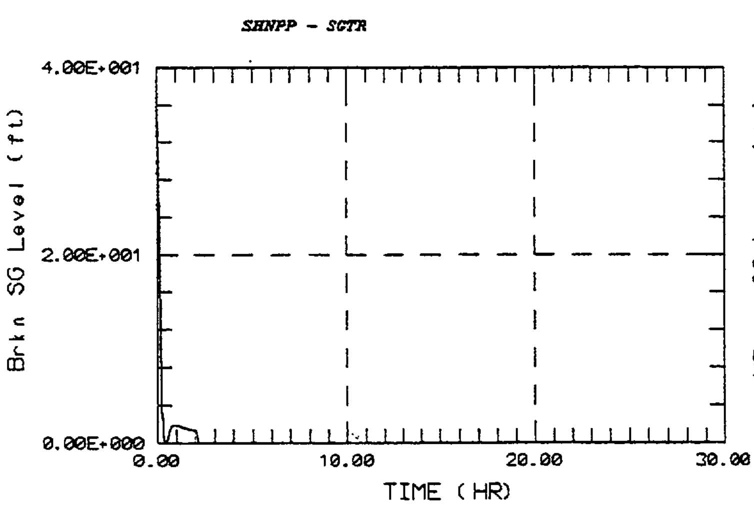

SHNPP - SGTR
RPV Water Level (in)
3.00E+001
2.00E+001
1.00E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

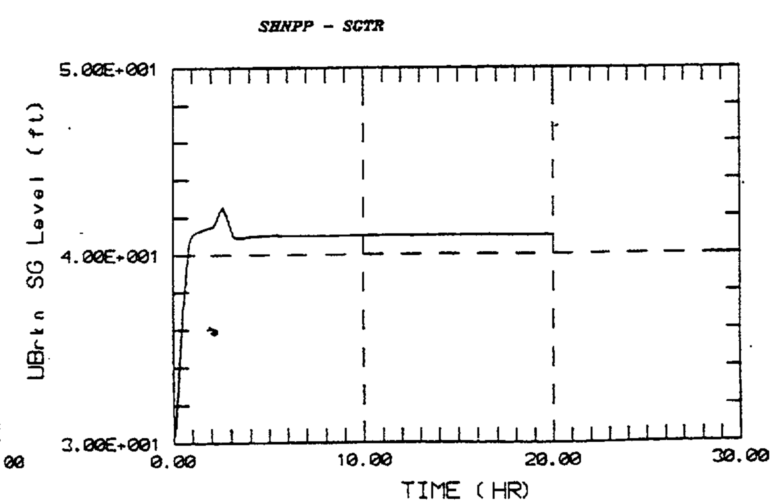

SHNPP - SGTR
RAB Node 2 Temp
9.00E+001
8.50E+001
8.00E+001
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - SGTR
RAB Node 3 Temp (F)
9.00E+001
8.50E+001
8.00E+001
0.00
10.00
20.00
30.00
TIME (HR)

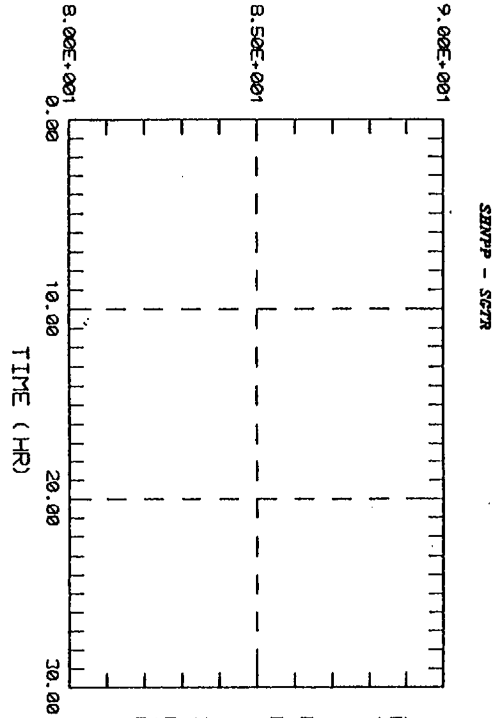

SHNPP - SGTR
t Press (psia)
3.00E+001
2.00E+001
1.00E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - SGTR
RAB Node 1 Temp (F)
9.00E+001
8.50E+001
8.00E+001
0.00
10.00
20.00
30.00
TIME (HR)

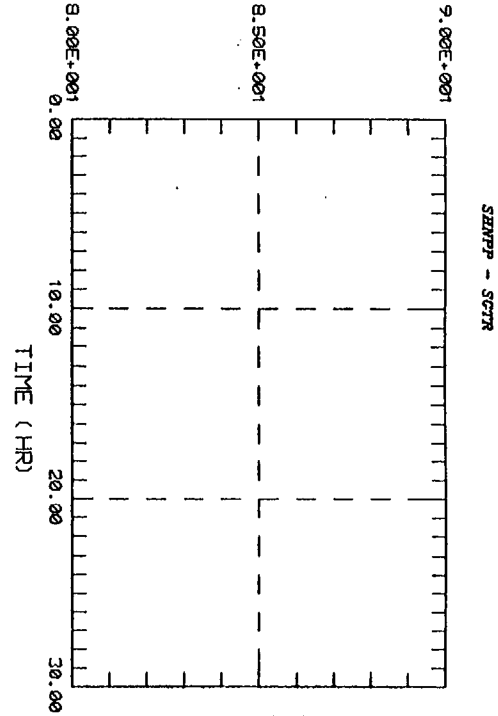

SHNPP - SGTR
Brkn SG Level (ft)
4.00E+001
2.00E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - SGTR
UBrkn SG Level (ft)
5.00E+001
4.00E+001
3.00E+001
0.00
10.00
20.00
30.00
TIME (HR)

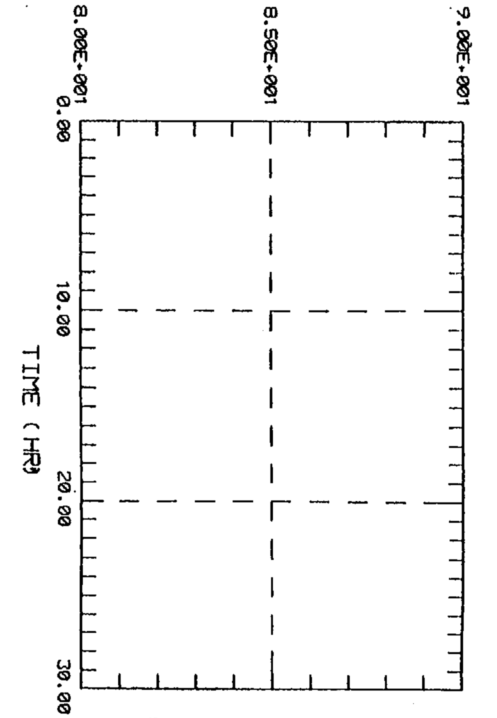
RAB Node 4 Temp (F)



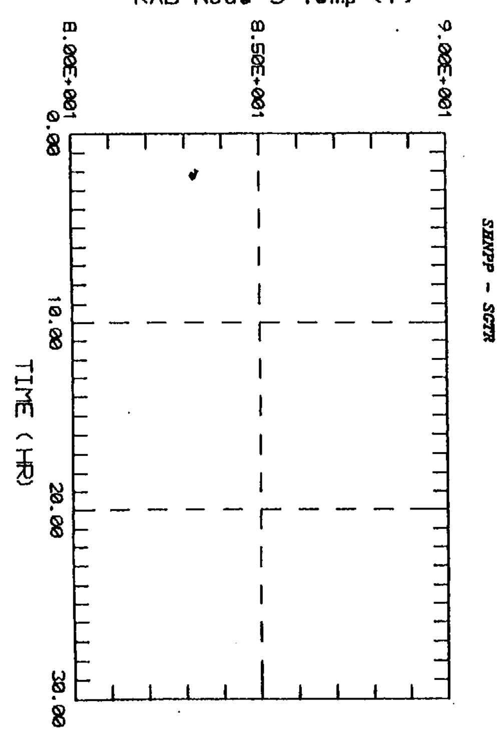
Node 6 Temp (F)



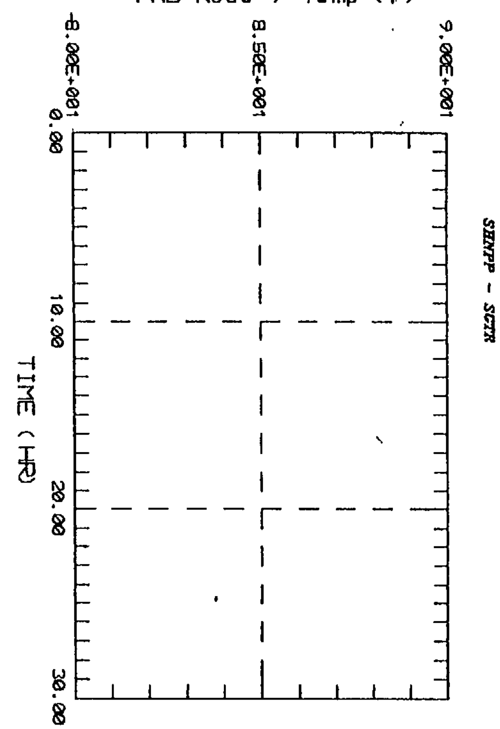
FHB Node 8 Temp



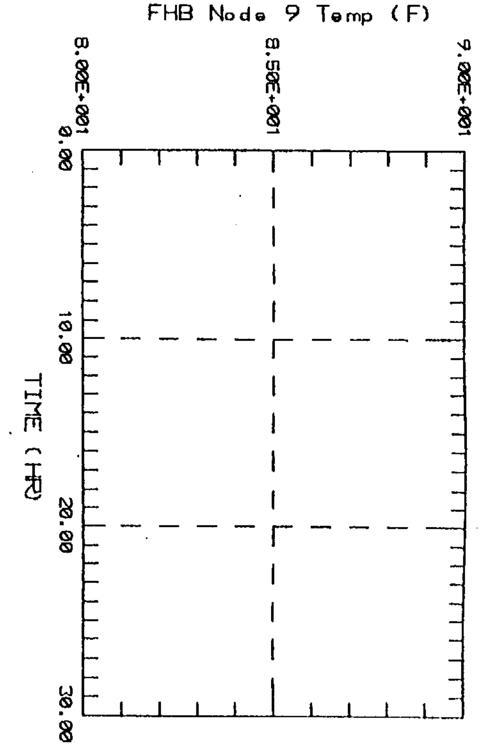
RAB Node 5 Temp (F)

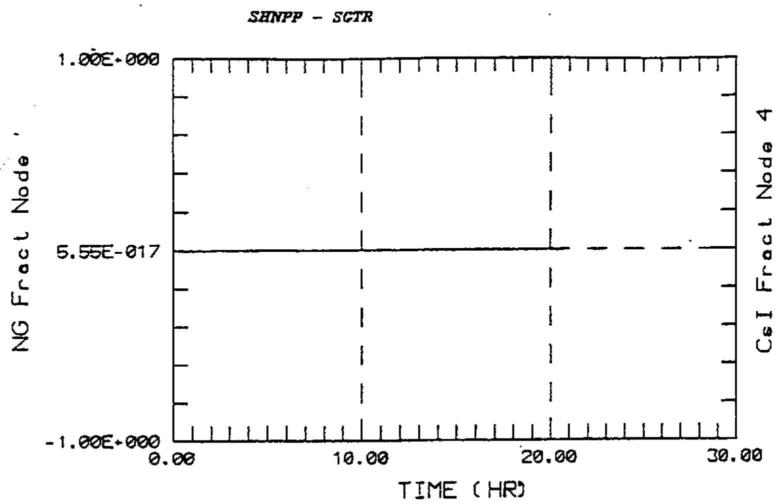


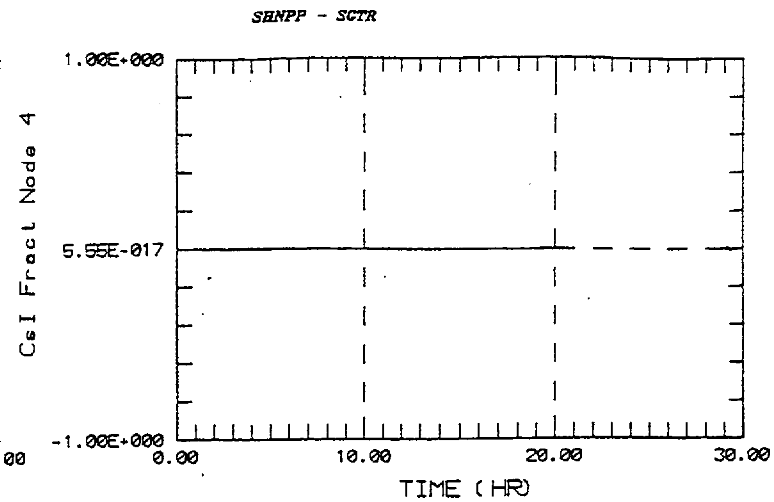
FHB Node 7 Temp (F)

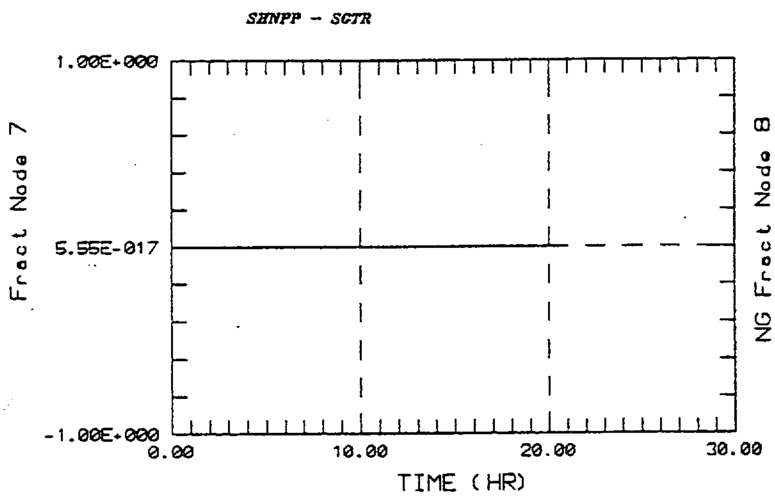


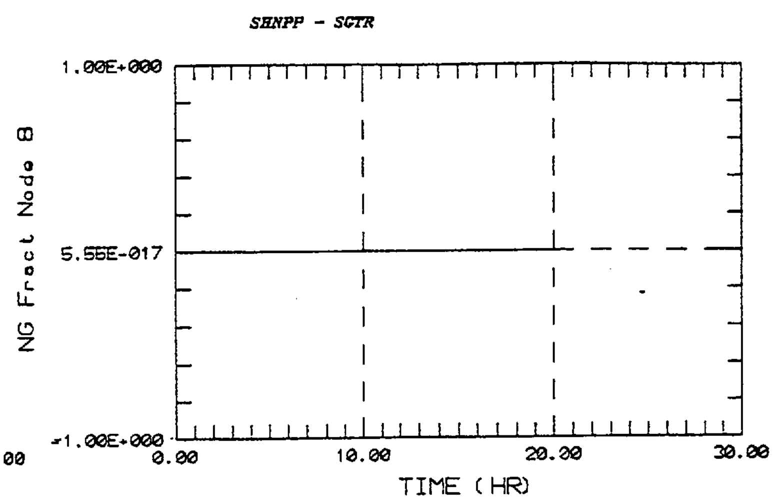
FHB Node 9 Temp (F)

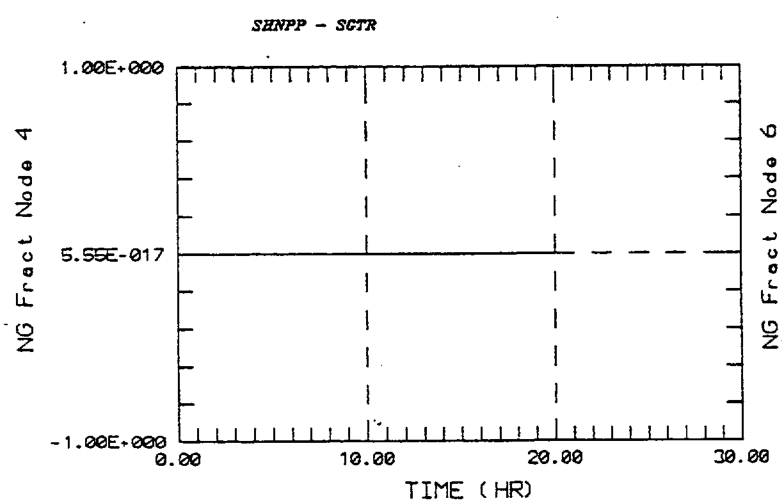


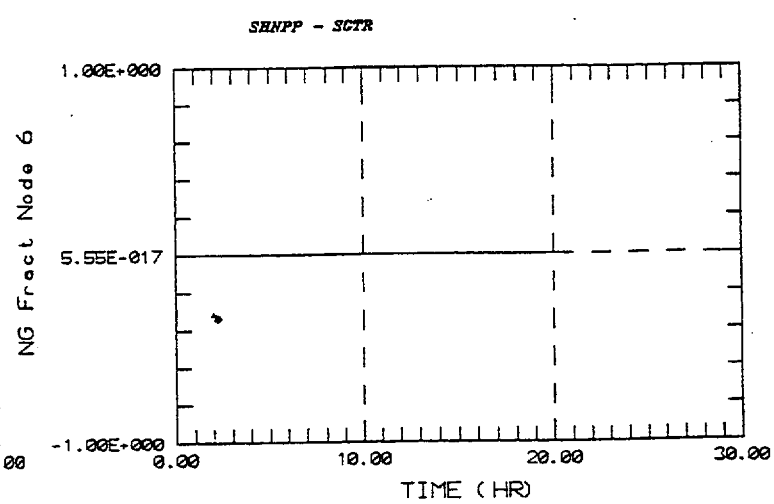

SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node
0.00
10.00
20.00
30.00
TIME (HR)

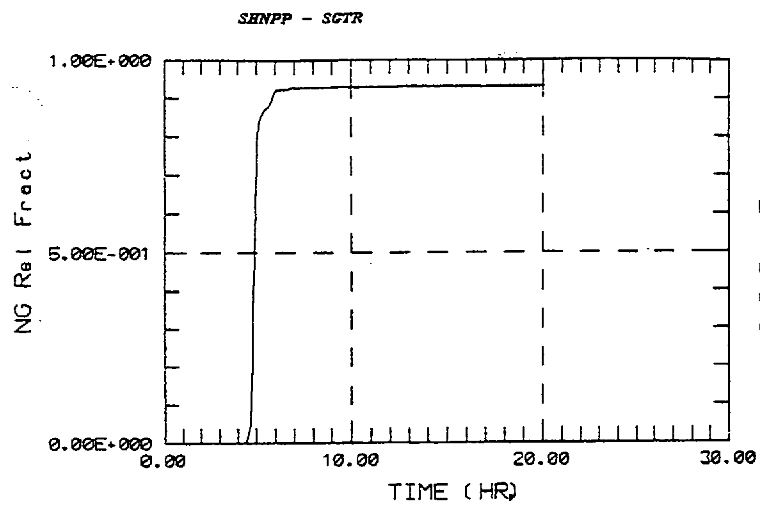

SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
CsI Fract Node 4
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
Fract Node 7
0.00
10.00
20.00
30.00
TIME (HR)

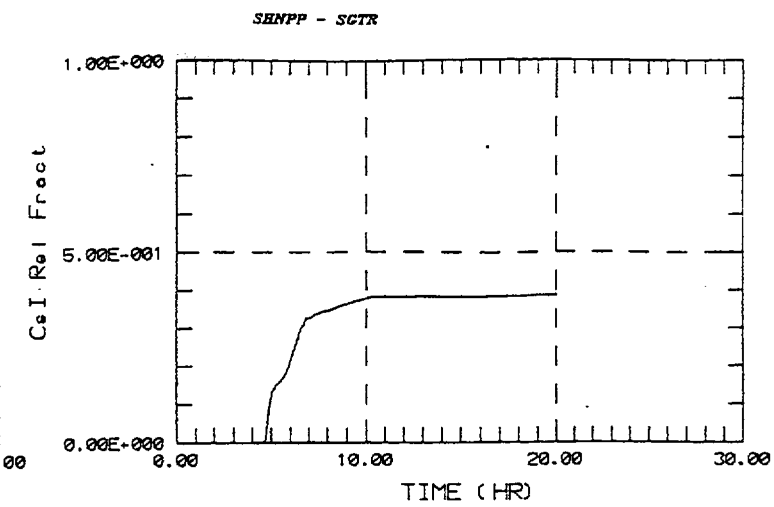

SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node 8
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node 4
0.00
10.00
20.00
30.00
TIME (HR)

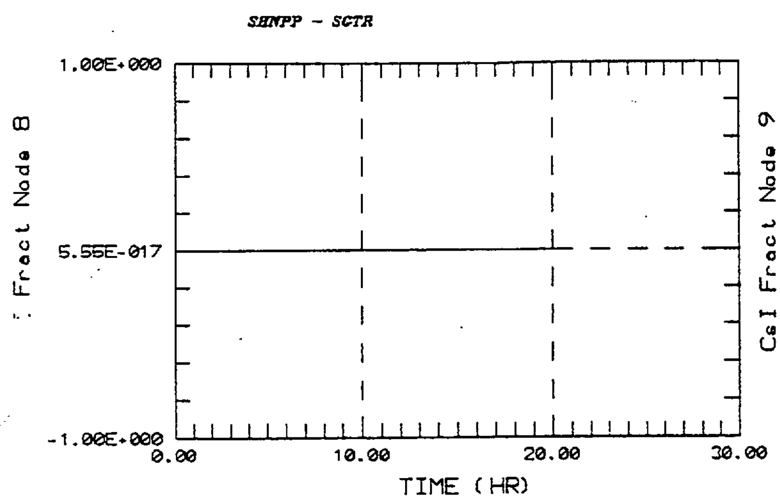

SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node 6
0.00
10.00
20.00
30.00
TIME (HR)



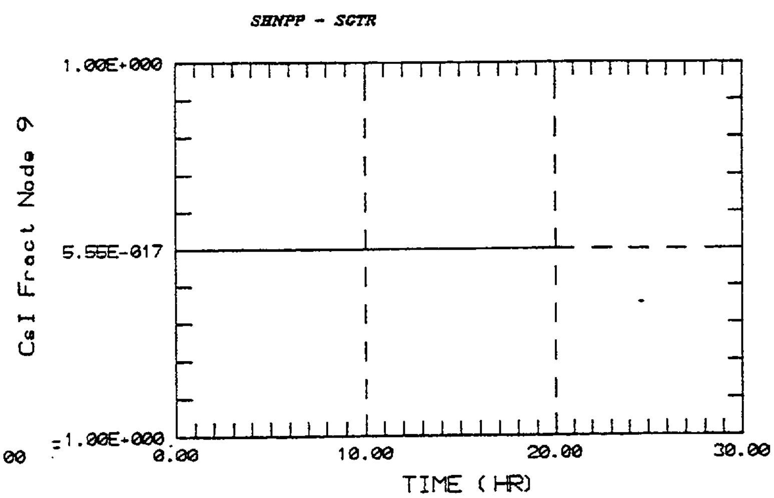
SHNPP - SGTR
1.00E+000
5.00E-001
0.00E+000
NG Rel Fract
0.00
10.00
20.00
30.00
TIME (HR)



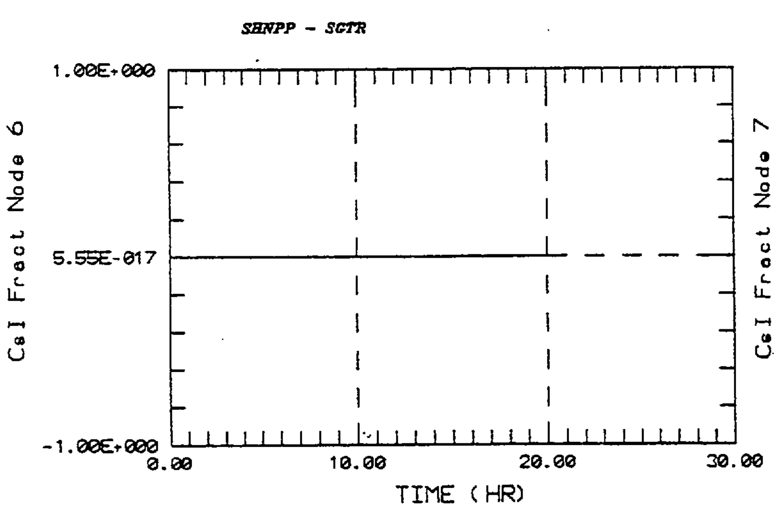
SHNPP - SGTR
1.00E+000
5.00E-001
0.00E+000
CsI Rel Fract
0.00
10.00
20.00
30.00
TIME (HR)



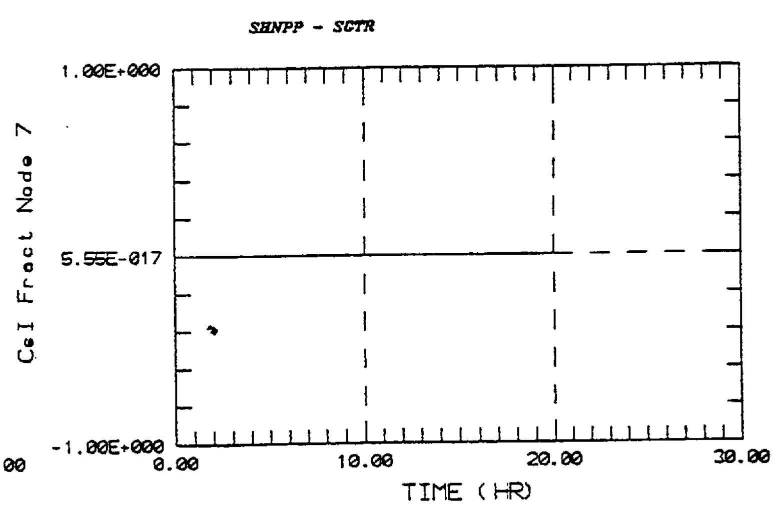
SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
Fract Node 8
0.00
10.00
20.00
30.00
TIME (HR)



SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
CsI Fract Node 9
0.00
10.00
20.00
30.00
TIME (HR)



SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
CsI Fract Node 6
0.00
10.00
20.00
30.00
TIME (HR)



SHNPP - SGTR
1.00E+000
5.55E-017
-1.00E+000
CsI Fract Node 7
0.00
10.00
20.00
30.00
TIME (HR)

```
C TEST
TITLE
        SHNPP - 5 inch Cont Iso Failure
        END TITLE

ATTACH ATTACH.SAM

PARAMETER CHANGES
        TDMAX   5  SECONDS
        PCF 145. PSI
        ZWSGL 29.7
        ZWCTLB 33. FT
        ZWCTLU 33. FT
        MWSG0 97000. LB
        WFWMX 1.493E5 LB/HR
C
        BREAK NODE 6
        BREAK AREA 0.000135 M**2
        BREAK ELEVATION 21. FT
        UNBROKEN BREAK NODE 12
        UNBROKEN BREAK AREA 0.00027 M**2
        UNBROKEN BREAK ELEVATION 21. FT
        20,3,0
        2 5.
        END OF PARAMETER CHANGES AND NOLIST

NOT A RESTART

PRINT TIME 6 HOURS

FINAL TIME 24.0

PARALLEL

WHEN BEGIN
     SCRAM ON
     MAIN COOLANT PUMPS OFF
     PZR HEATERS OFF
     PZR SPRAYS OFF
     HPI OFF
     LPI OFF
     CON SPRAYS OFF
     FAN COOLERS OFF
     MAKEUP OFF
     LETDOWN OFF
     PARAMETER CHANGES
         ACFPR  .1364 DT**2
         PCF 14.7 PSI
         END
     END

WHEN TIME > 1.5 HOURS
     BREAK ON
     END

WHEN TIME > 4.0 HOURS
     AFW OFF
```

```
     END

INTERVENTION 49
WHEN RPV FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 50
WHEN CONTMT FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 51
WHEN CORE UNCOVERED IS TRUE
LET CORE UNC TIME = TIME
END

INTERVENTION 52
WHEN HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
  OR HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
LET MELT ONSET TIME = TIME
END

INTERVENTION 53
WHEN PEAK CORE TEMP > TCRHOT
LET PEAK CORE TIME = TIME
RESTORE
SILENT
END

INTERVENTION 54
WHEN PEAK CONTMT PRESSURE = PA
LET PEAK PA TIME = TIME
RESTORE
SILENT
END

INTERVENTION 55
WHEN IEVNT(103) IS TRUE
 OR IEVNT(79) IS TRUE
 PARAMETER CHANGE
  FLPHI  10.0
END
RESTORE 56
END

INTERVENTION 56
WHEN IEVNT(103) IS FALSE
 AND IEVNT(79) IS FALSE
 PARAMETER CHANGE
  FLPHI  2.0
END
RESTORE 55
END
```

```
INTERVENTION 57
WHEN IEVNT(42) IS TRUE
 INCREMENT PZR SVS OPEN BY 1
RESTORE 58
END

INTERVENTION 58
WHEN IEVNT(42) IS FALSE
RESTORE 57
END

WHEN PA > 99.7 PSI
LET CONTMT OVERPRESS TIME = TIME
FULL OUTPUT
REPORT
END
```

Containment Isolation Failure

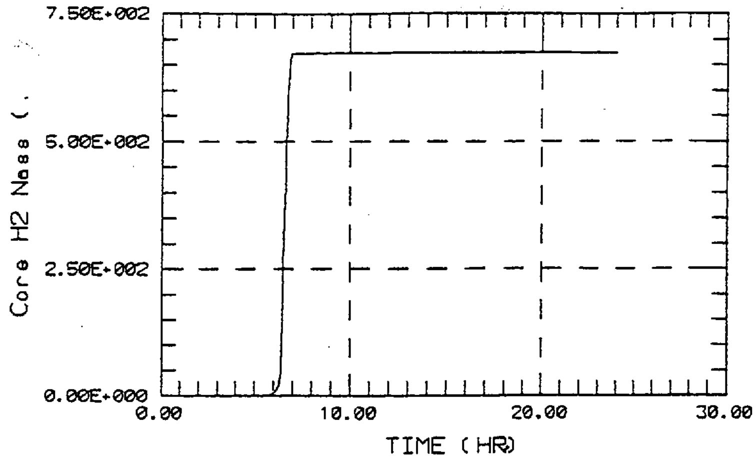
0.0	4	MAIN COOLANT PUMPS OFF
0.0	13	REACTOR SCRAM
0.0	46	LETDOWN FLOW OFF
0.0	154	AUX FEEDWATER ON
0.0	156	MSIV CLOSED
0.0	178	AUX CO2 SUPLY DEPLETD
0.0	190	UHI ACCUM EMPTY
0.0	215	MCP SWITCH OFF OR HI-VIBR TRIP
0.0	216	HPI FORCED OFF
0.0	217	LPI TRAIN 1 FORCED OFF
0.0	221	FANS/COOLERS FORCED OFF
0.0	222	CONTMT SPRAYS FORCED OFF
0.0	223	PZR SPRAYS FORCED OFF
0.0	226	1 PZR HTRS FORCED OFF
0.0	227	MANUAL SCRAM
0.0	242	PS MAKEUP OFF
0.0	243	LETDOWN SWITCH OFF
2.0	104	CONTMT FAILED
5.2	162	SEC RV OPEN UNBROKEN S/G'S
6.2	152	SEC RV OPEN BROKEN S/G
20.2	152	SEC RV NOT OPEN BROKEN S/G
20.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
21.9	152	SEC RV OPEN BROKEN S/G
21.9	162	SEC RV OPEN UNBROKEN S/G'S
23.5	152	SEC RV NOT OPEN BROKEN S/G
23.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
25.2	152	SEC RV OPEN BROKEN S/G
25.2	162	SEC RV OPEN UNBROKEN S/G'S
26.9	152	SEC RV NOT OPEN BROKEN S/G
26.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
28.5	152	SEC RV OPEN BROKEN S/G
28.5	162	SEC RV OPEN UNBROKEN S/G'S
30.2	152	SEC RV NOT OPEN BROKEN S/G
30.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
44.4	152	SEC RV OPEN BROKEN S/G
44.4	162	SEC RV OPEN UNBROKEN S/G'S
46.1	152	SEC RV NOT OPEN BROKEN S/G
46.1	162	SEC RV NOT OPEN UNBROKEN S/G'S
75.5	152	(20) SEC RV NOT OPEN BROKEN S/G
82.3	162	(20) SEC RV NOT OPEN UNBROKEN S/G'S
161.4	152	(30) SEC RV NOT OPEN BROKEN S/G
161.4	162	(30) SEC RV NOT OPEN UNBROKEN S/G'S
204.7	152	(40) SEC RV NOT OPEN BROKEN S/G
211.5	162	(40) SEC RV NOT OPEN UNBROKEN S/G'S
278.7	152	(50) SEC RV NOT OPEN BROKEN S/G
282.1	162	(50) SEC RV NOT OPEN UNBROKEN S/G'S
353.1	152	(60) SEC RV NOT OPEN BROKEN S/G
353.1	162	(60) SEC RV NOT OPEN UNBROKEN S/G'S
406.4	152	(70) SEC RV NOT OPEN BROKEN S/G
424.1	162	(70) SEC RV NOT OPEN UNBROKEN S/G'S
481.0	152	(80) SEC RV NOT OPEN BROKEN S/G
491.2	162	(80) SEC RV NOT OPEN UNBROKEN S/G'S
545.8	152	(90) SEC RV NOT OPEN BROKEN S/G
555.9	162	(90) SEC RV NOT OPEN UNBROKEN S/G'S
611.1	152	(100) SEC RV NOT OPEN BROKEN S/G

626.3	162	(100) SEC RV NOT OPEN UNBROKEN S/G'S
958.2	152	(150) SEC RV NOT OPEN BROKEN S/G
968.3	162	(150) SEC RV NOT OPEN UNBROKEN S/G'S
1358.9	152	(200) SEC RV NOT OPEN BROKEN S/G
1374.0	162	(200) SEC RV NOT OPEN UNBROKEN S/G'S
1771.0	152	(250) SEC RV NOT OPEN BROKEN S/G
1786.2	162	(250) SEC RV NOT OPEN UNBROKEN S/G'S
2246.5	152	(300) SEC RV NOT OPEN BROKEN S/G
2261.6	162	(300) SEC RV NOT OPEN UNBROKEN S/G'S
2723.2	152	(350) SEC RV NOT OPEN BROKEN S/G
2743.4	162	(350) SEC RV NOT OPEN UNBROKEN S/G'S
3263.4	152	(400) SEC RV NOT OPEN BROKEN S/G
3288.5	162	(400) SEC RV NOT OPEN UNBROKEN S/G'S
3804.9	152	(450) SEC RV NOT OPEN BROKEN S/G
3825.1	162	(450) SEC RV NOT OPEN UNBROKEN S/G'S
4409.9	152	(500) SEC RV NOT OPEN BROKEN S/G
4430.0	162	(500) SEC RV NOT OPEN UNBROKEN S/G'S
5400.0	209	PS BREAK(S) FAILED
5405.9	81	WATER ON LOWER CMPT FLOOR
5472.6	14	FP MODELS ON
5921.2	32	PZR EMPTY
6001.5	32	PZR NOT EMPTY
6006.5	32	PZR EMPTY
6085.3	32	PZR NOT EMPTY
6102.0	32	PZR EMPTY
8136.7	32	PZR NOT EMPTY
10157.3	57	WATER IN CAVITY
11755.4	152	(1000) SEC RV NOT OPEN BROKEN S/G
11785.5	162	(1000) SEC RV NOT OPEN UNBROKEN S/G'S
14400.0	154	AUX FEEDWATER OFF
14400.0	224	AUX FEED WATER FORCED OFF
17399.9	32	PZR EMPTY
18700.2	152	(1500) SEC RV NOT OPEN BROKEN S/G
18761.5	25	PS NONEQ THERMO
18761.5	162	(1500) SEC RV NOT OPEN UNBROKEN S/G'S
18897.3	49	CORE HAS UNCOV
24444.2	152	SEC RV OPEN BROKEN S/G
24444.4	152	SEC RV NOT OPEN BROKEN S/G
24491.9	152	SEC RV OPEN BROKEN S/G
24493.2	152	SEC RV NOT OPEN BROKEN S/G
24779.8	152	SEC RV OPEN BROKEN S/G
24782.0	152	SEC RV NOT OPEN BROKEN S/G
25060.1	152	SEC RV OPEN BROKEN S/G
25062.3	152	SEC RV NOT OPEN BROKEN S/G
25252.3	152	SEC RV OPEN BROKEN S/G
25254.4	152	SEC RV NOT OPEN BROKEN S/G
25702.4	152	(20) SEC RV NOT OPEN BROKEN S/G
27432.0	152	(30) SEC RV NOT OPEN BROKEN S/G
28623.9	2	SUPPORT PLATE FAILED
28642.5	152	(40) SEC RV NOT OPEN BROKEN S/G
28683.9	3	RV FAILED
28684.0	61	CORIUM IN CAVITY
28684.2	69	WATER-CORIUM INTERACTION HAS OCCURED IN CAVITY
28684.2	59	WATER FLOODING IN CAVITY TO B
28684.2	59	WATER NOT FLOODING IN CAVITY TO B
28685.7	27	UNBKN LOOPS NOT BLOCKED AT PUMP BOWLS
28688.4	59	WATER FLOODING IN CAVITY TO B

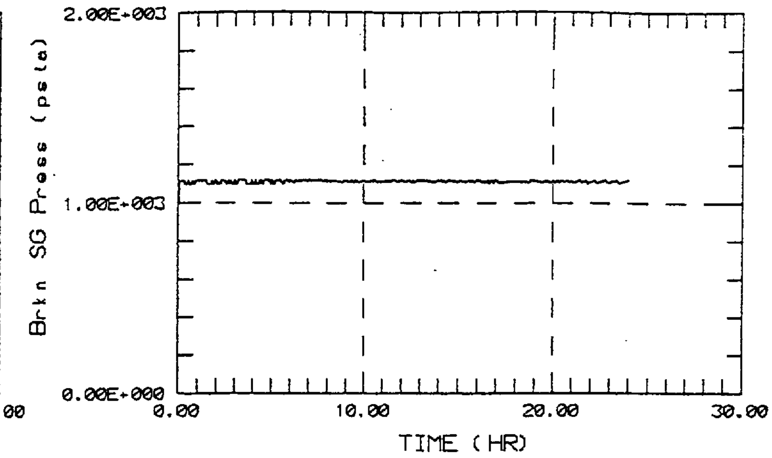
28688.4	58	CORIUM FLOODING IN CAVITY TO B
28688.4	58	CORIUM NOT FLOODING IN CAVITY TO B
28688.4	82	CORIUM IN LOWER CMPT
28688.5	58	CORIUM FLOODING IN CAVITY TO B
28688.6	75	BURN IN PROGRESS IN LOWER CMPT
28690.2	28	DWNCMR NOT BLCKD FOR GAS XPORT
28691.6	58	CORIUM NOT FLOODING IN CAVITY TO B
28691.7	58	CORIUM FLOODING IN CAVITY TO B
28691.7	58	CORIUM NOT FLOODING IN CAVITY TO B
28691.7	58	CORIUM FLOODING IN CAVITY TO B
28691.8	58	CORIUM NOT FLOODING IN CAVITY TO B
28697.1	59	WATER NOT FLOODING IN CAVITY TO B
28697.2	59	WATER FLOODING IN CAVITY TO B
28697.3	59	WATER NOT FLOODING IN CAVITY TO B
28703.0	75	NO BURN IN LOWER CMPT
28703.4	59	WATER FLOODING IN CAVITY TO B
28703.4	59	WATER NOT FLOODING IN CAVITY TO B
28717.5	59	WATER FLOODING IN CAVITY TO B
28717.5	59	WATER NOT FLOODING IN CAVITY TO B
28717.5	188	ACCUMULATOR WATER DEPLETED
30053.8	152	(50)SEC RV NOT OPEN BROKEN S/G
31727.2	152	(60)SEC RV NOT OPEN BROKEN S/G
33630.6	152	(70)SEC RV NOT OPEN BROKEN S/G
35998.1	152	(80)SEC RV NOT OPEN BROKEN S/G
38654.9	152	(90)SEC RV NOT OPEN BROKEN S/G
41539.1	152	(100)SEC RV NOT OPEN BROKEN S/G
42093.5	163	SEC SV(S) OPEN UNBROKEN S/G'S
42098.0	163	SEC SV(S) NOT OPEN UNBROKEN S/G'S
42099.0	163	SEC SV(S) OPEN UNBROKEN S/G'S
42100.0	163	SEC SV(S) NOT OPEN UNBROKEN S/G'S
42101.0	163	SEC SV(S) OPEN UNBROKEN S/G'S
42102.0	163	SEC SV(S) NOT OPEN UNBROKEN S/G'S
42913.9	161	UNBKN S/G DRY
43315.9	162	SEC RV OPEN UNBROKEN S/G'S
43316.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
43502.9	162	SEC RV OPEN UNBROKEN S/G'S
43503.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
43511.9	162	SEC RV OPEN UNBROKEN S/G'S
43512.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
43527.9	162	SEC RV OPEN UNBROKEN S/G'S
43528.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
43729.9	162	SEC RV OPEN UNBROKEN S/G'S
43730.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
46863.8	162	SEC RV OPEN UNBROKEN S/G'S
46865.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
48298.5	162	SEC RV OPEN UNBROKEN S/G'S
48300.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
49859.3	162	SEC RV OPEN UNBROKEN S/G'S
49861.3	162	SEC RV NOT OPEN UNBROKEN S/G'S
50471.1	81	LOWER CMPT FLOOR DRY
51565.1	162	SEC RV OPEN UNBROKEN S/G'S
51567.0	162	SEC RV NOT OPEN UNBROKEN S/G'S
53424.8	162	SEC RV OPEN UNBROKEN S/G'S
53426.7	162	SEC RV NOT OPEN UNBROKEN S/G'S
53814.7	152	(150)SEC RV NOT OPEN BROKEN S/G
55374.6	162	SEC RV OPEN UNBROKEN S/G'S
55376.5	162	SEC RV NOT OPEN UNBROKEN S/G'S

57405.3	162	SEC RV OPEN UNBROKEN S/G'S
57407.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
59505.1	162	SEC RV OPEN UNBROKEN S/G'S
59506.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
61683.8	162	SEC RV OPEN UNBROKEN S/G'S
61685.7	162	SEC RV NOT OPEN UNBROKEN S/G'S
63943.8	162	SEC RV OPEN UNBROKEN S/G'S
63945.7	162	SEC RV NOT OPEN UNBROKEN S/G'S
66294.4	162	SEC RV OPEN UNBROKEN S/G'S
66296.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
68749.4	162	SEC RV OPEN UNBROKEN S/G'S
68751.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
69893.4	152	(200)SEC RV NOT OPEN BROKEN S/G
71309.8	162	SEC RV OPEN UNBROKEN S/G'S
71311.6	162	SEC RV NOT OPEN UNBROKEN S/G'S
73989.3	162	SEC RV OPEN UNBROKEN S/G'S
73991.0	162	SEC RV NOT OPEN UNBROKEN S/G'S
74990.5	151	BROKEN S/G DRY
75900.5	152	SEC RV OPEN BROKEN S/G
75902.8	152	SEC RV NOT OPEN BROKEN S/G
76799.7	162	SEC RV OPEN UNBROKEN S/G'S
76801.4	162	SEC RV NOT OPEN UNBROKEN S/G'S
77131.6	152	SEC RV OPEN BROKEN S/G
77133.9	152	SEC RV NOT OPEN BROKEN S/G
78595.7	152	SEC RV OPEN BROKEN S/G
78598.0	152	SEC RV NOT OPEN BROKEN S/G
79759.7	162	SEC RV OPEN UNBROKEN S/G'S
79761.4	162	SEC RV NOT OPEN UNBROKEN S/G'S
80326.6	152	SEC RV OPEN BROKEN S/G
80328.8	152	SEC RV NOT OPEN BROKEN S/G
82300.5	152	SEC RV OPEN BROKEN S/G
82302.6	152	SEC RV NOT OPEN BROKEN S/G
82879.2	162	SEC RV OPEN UNBROKEN S/G'S
82880.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
84446.1	152	SEC RV OPEN BROKEN S/G
84448.2	152	SEC RV NOT OPEN BROKEN S/G
86184.7	162	SEC RV OPEN UNBROKEN S/G'S
86186.4	162	SEC RV NOT OPEN UNBROKEN S/G'S

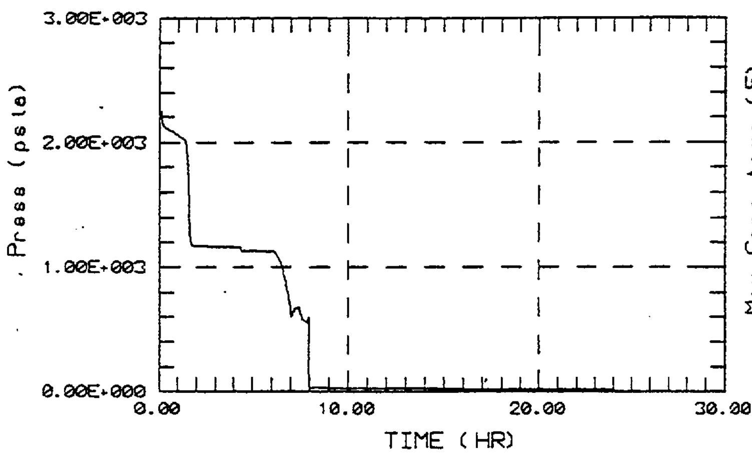
SHNPP – 5 inch Cont Iso Failure



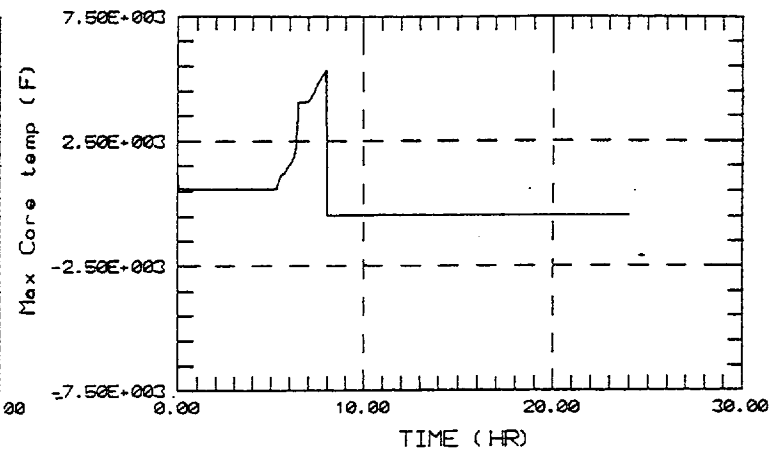
SHNPP – 5 inch Cont Iso Failure



SHNPP – 5 inch Cont Iso Failure



SHNPP – 5 inch Cont Iso Failure



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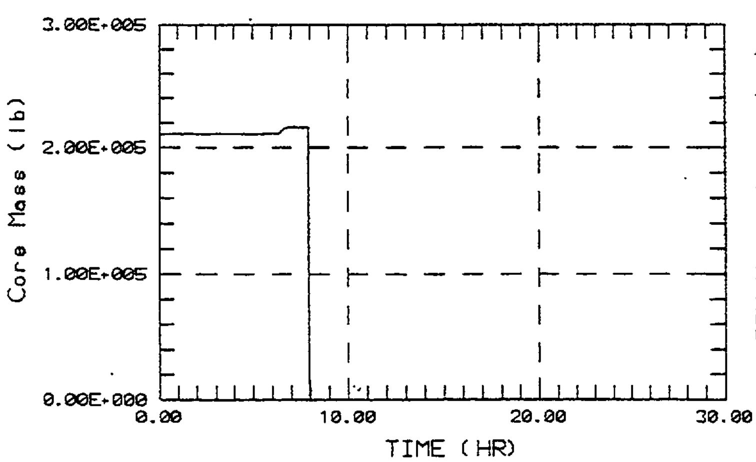
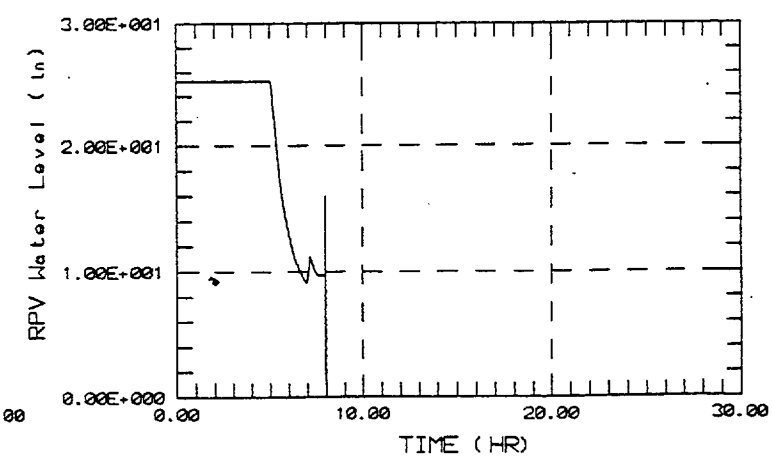
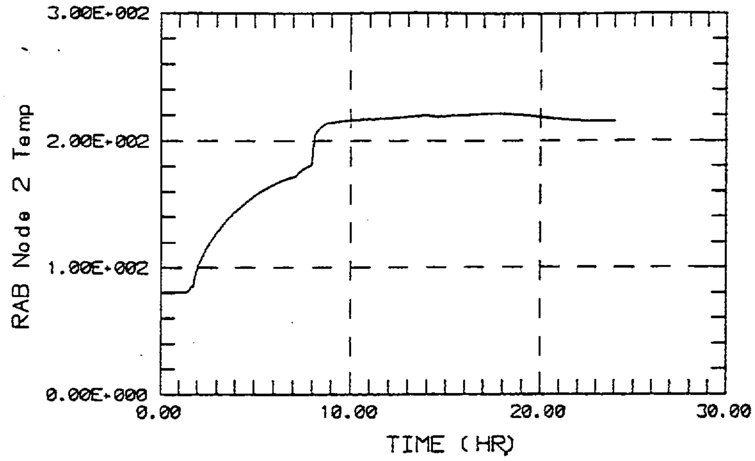


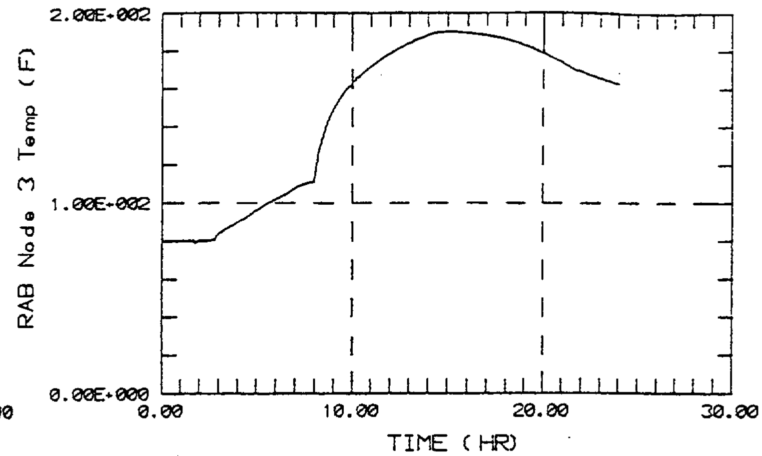
SHNPP – 5 inch Cont Iso Failure



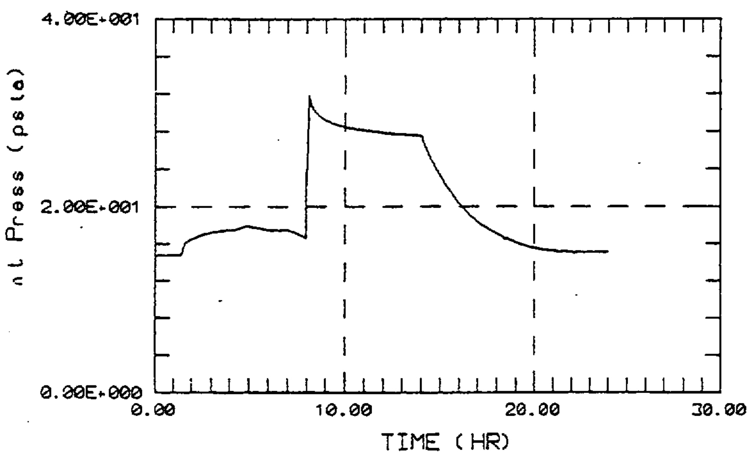
SHNPP - 5 inch Cont Iso Failure



SHNPP - 5 inch Cont Iso Failure



SHNPP - 5 inch Cont Iso Failure



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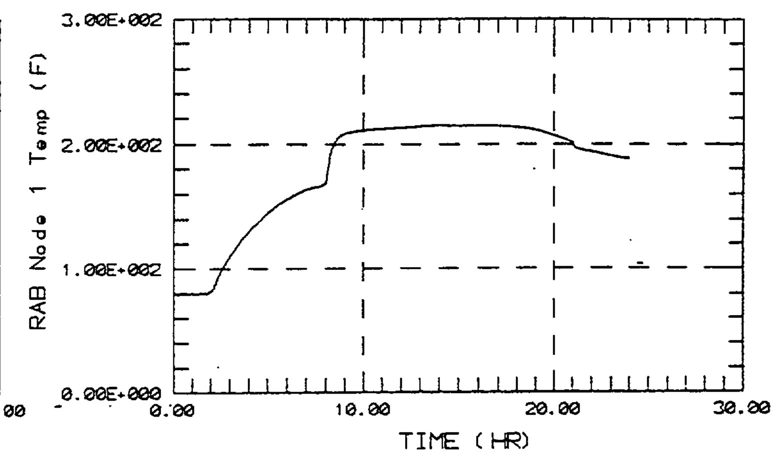
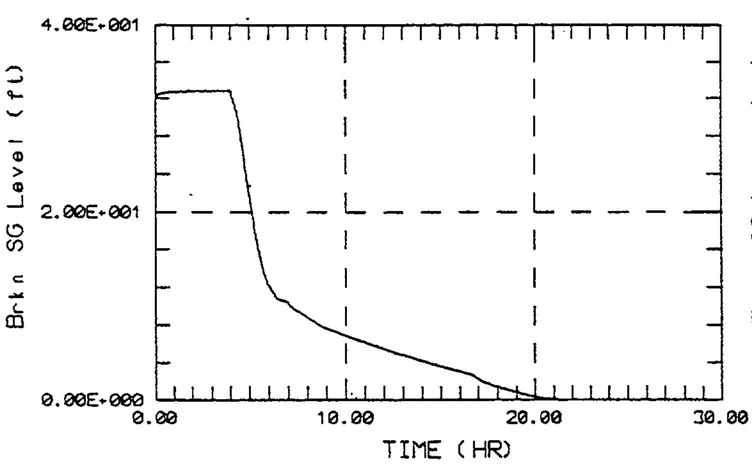
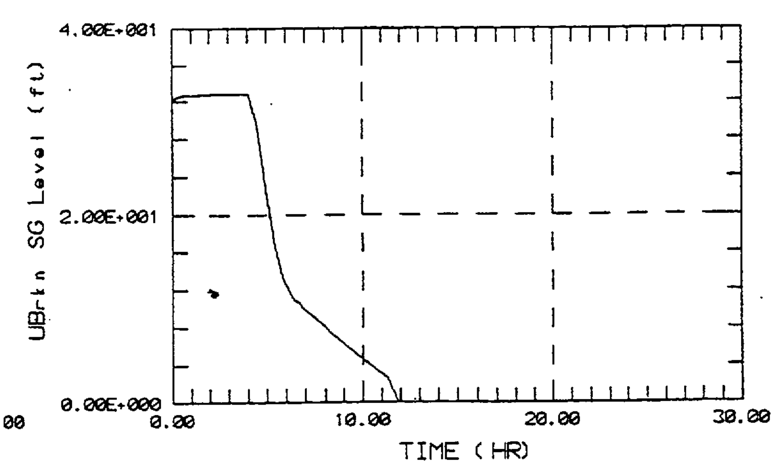


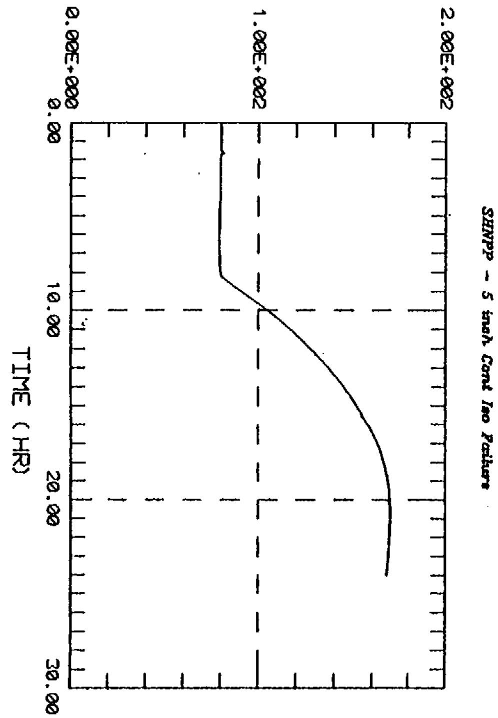
SHNPP - 5 inch Cont Iso Failure



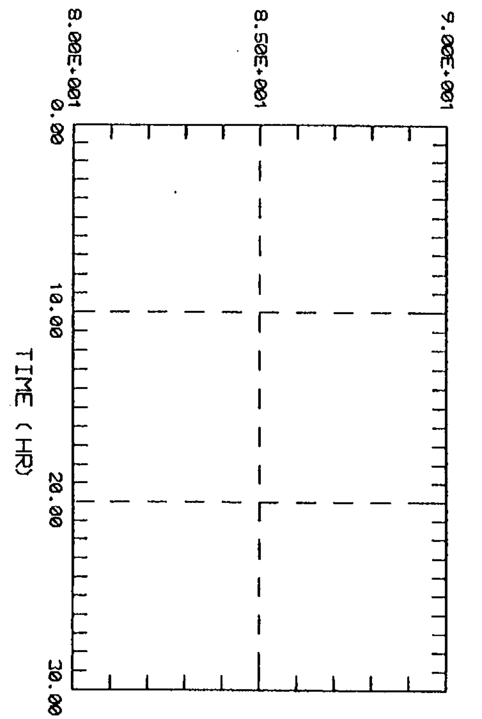
SHNPP - 5 inch Cont Iso Failure



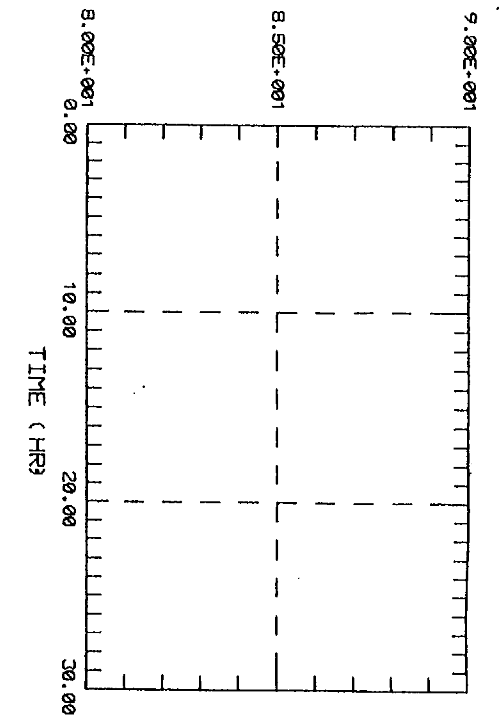
RAB Node 4 Temp (F)



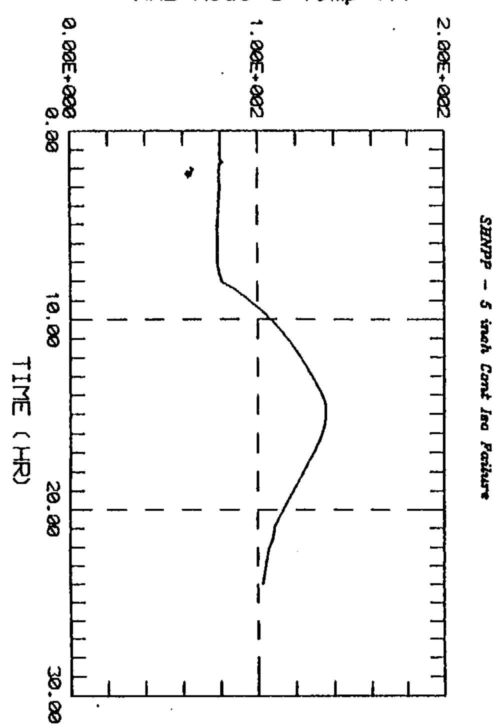
Node 6 Temp (F)



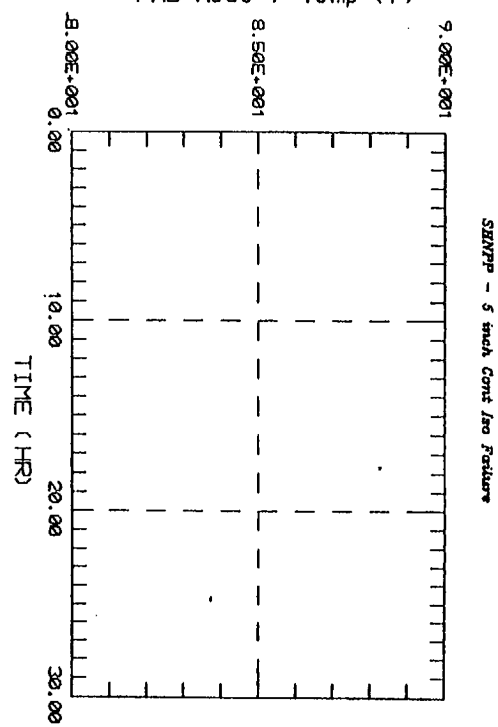
FHB Node 8 Temp



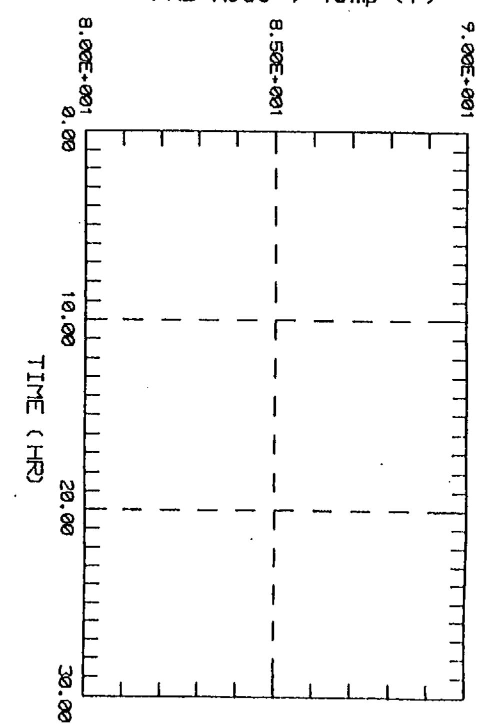
RAB Node 5 Temp (F)

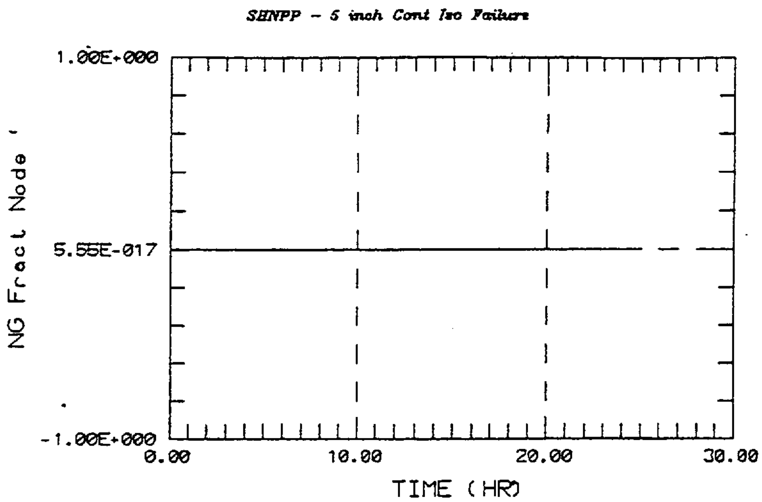


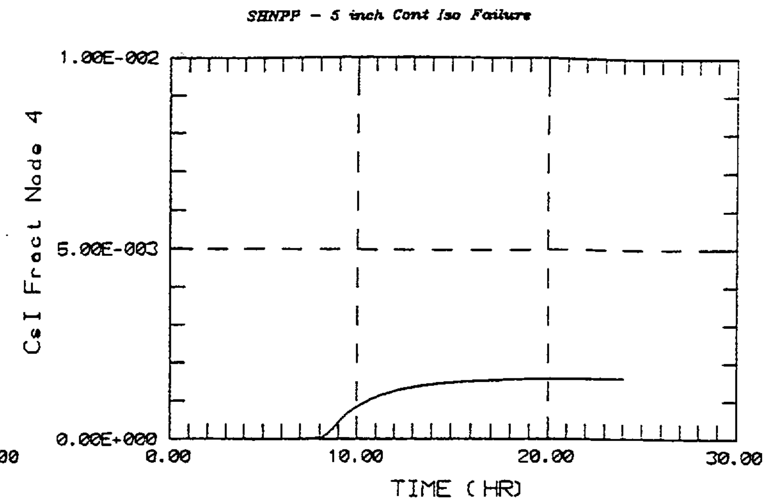
FHB Node 7 Temp (F)

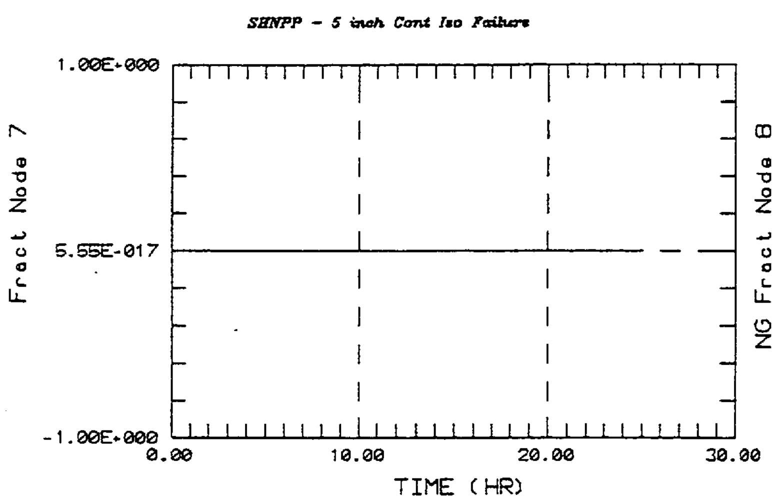


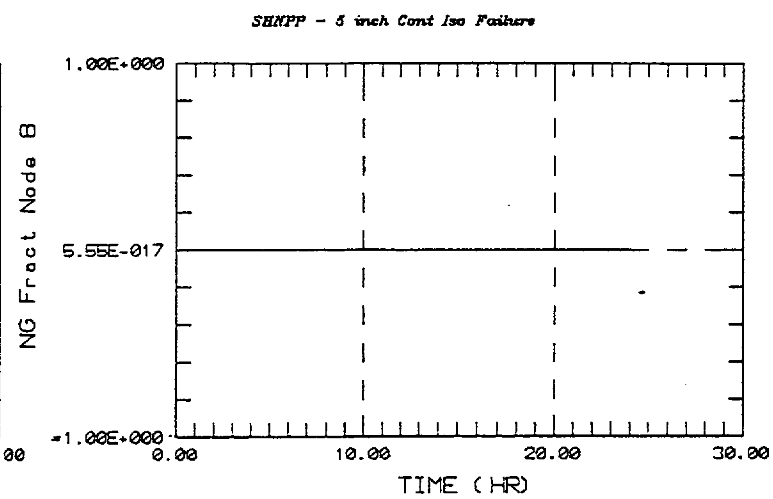
FHB Node 9 Temp (F)

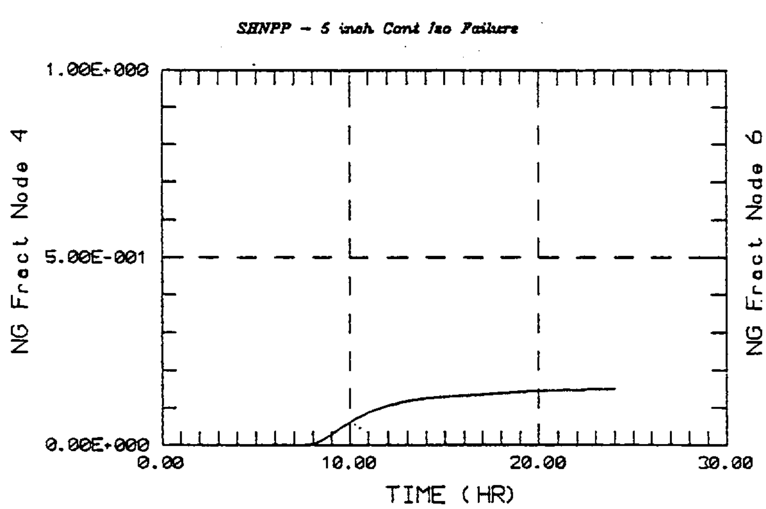


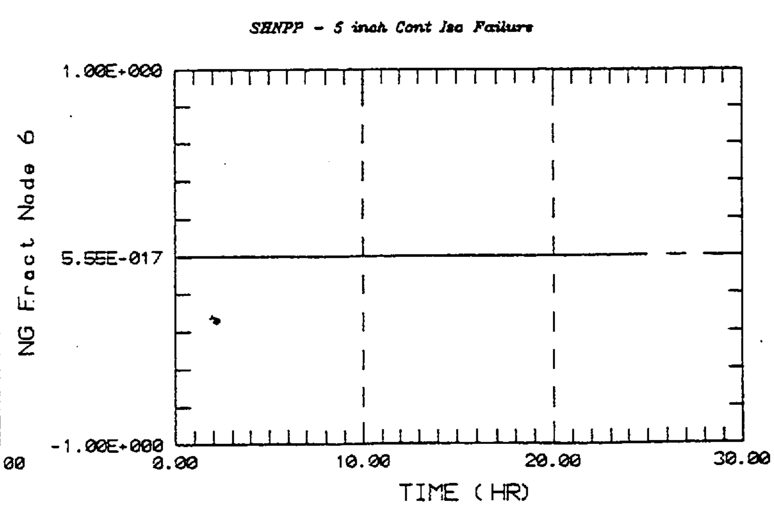

SHNPP - 5 inch Cont Iso Failure
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node
0.00
10.00
20.00
30.00
TIME (HR)

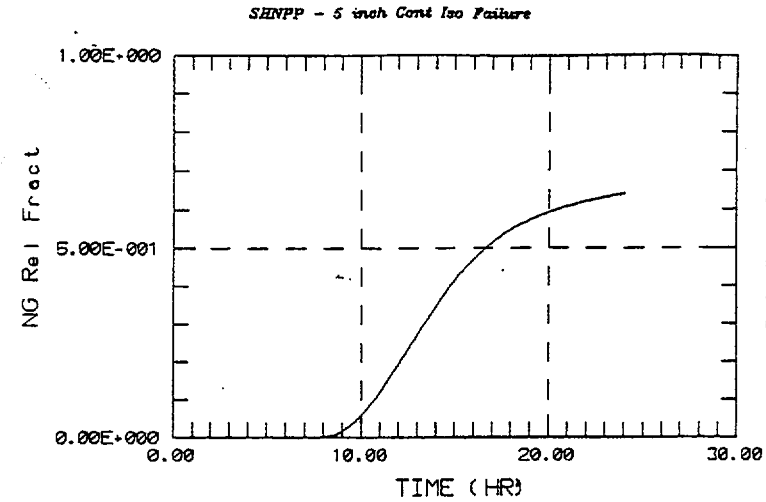

SHNPP - 5 inch Cont Iso Failure
1.00E-002
5.00E-003
0.00E+000
CsI Fract Node 4
0.00
10.00
20.00
30.00
TIME (HR)

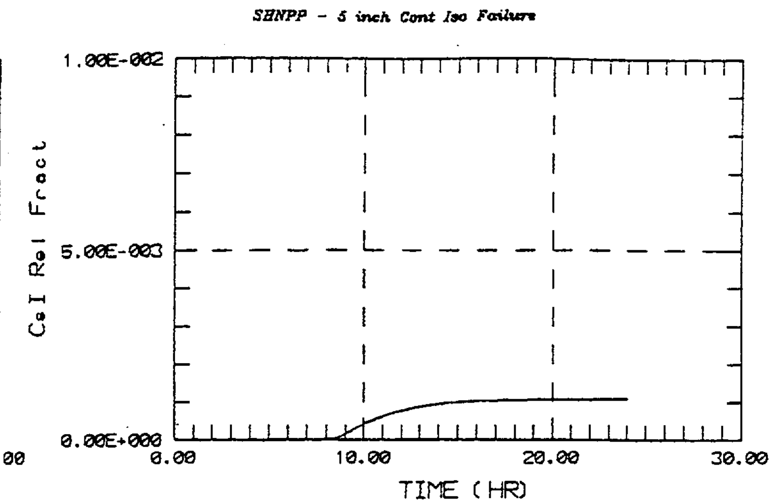

SHNPP - 5 inch Cont Iso Failure
1.00E+000
5.55E-017
-1.00E+000
Fract Node 7
0.00
10.00
20.00
30.00
TIME (HR)

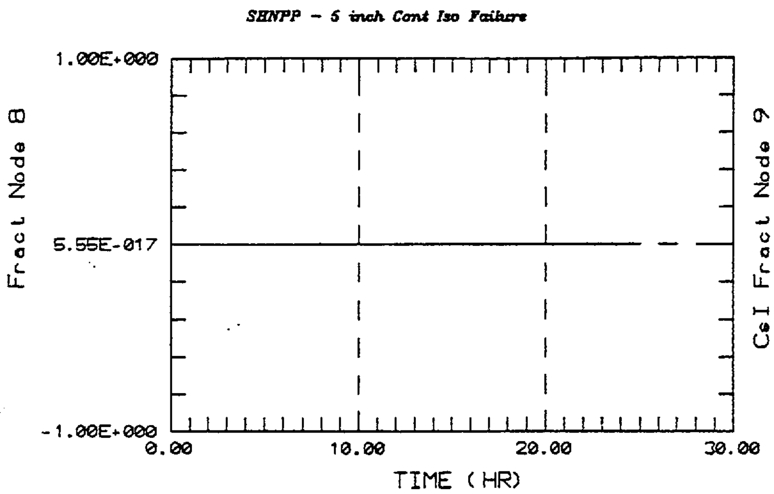

SHNPP - 5 inch Cont Iso Failure
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node 8
0.00
10.00
20.00
30.00
TIME (HR)

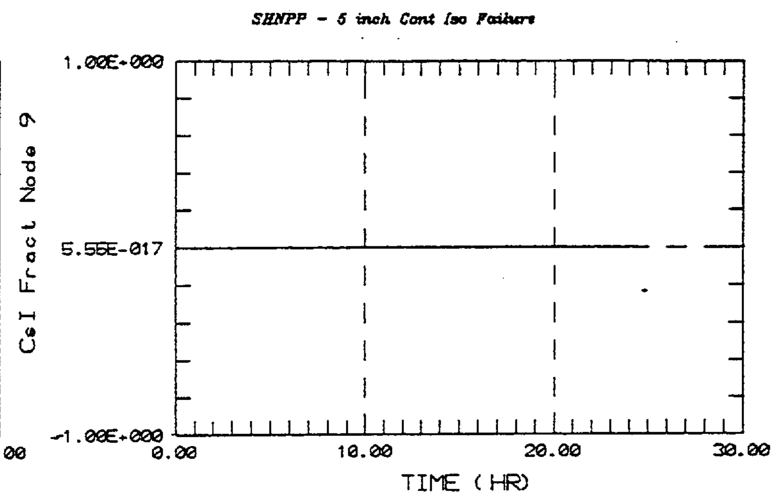

SHNPP - 5 inch Cont Iso Failure
1.00E+000
5.00E-001
0.00E+000
NG Fract Node 4
0.00
10.00
20.00
30.00
TIME (HR)

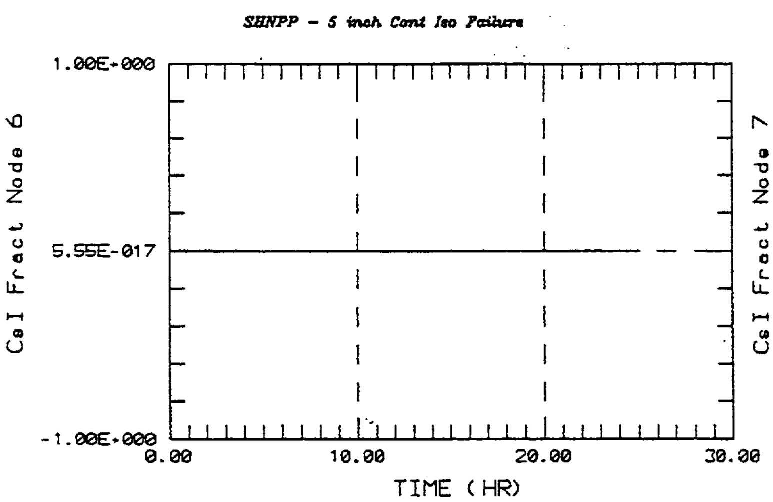

SHNPP - 5 inch Cont Iso Failure
1.00E+000
5.55E-017
-1.00E+000
NG Fract Node 6
0.00
10.00
20.00
30.00
TIME (HR)

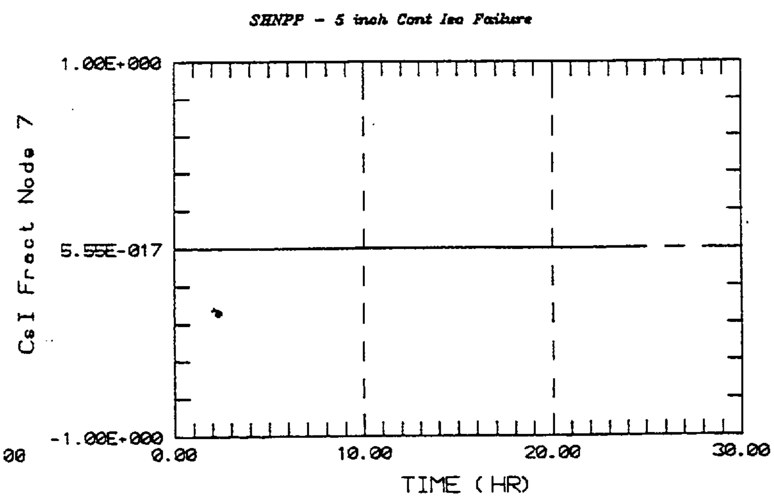

SHNPP - 5 inch Cont Iso Failure
NG Rel Fract
1.00E+000
5.00E-001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - 5 inch Cont Iso Failure
CsI Rel Fract
1.00E-002
5.00E-003
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - 5 inch Cont Iso Failure
Fract Node 8
1.00E+000
5.55E-017
-1.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - 5 inch Cont Iso Failure
CsI Fract Node 9
1.00E+000
5.55E-017
-1.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - 5 inch Cont Iso Failure
CsI Fract Node 6
1.00E+000
5.55E-017
-1.00E+000
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - 5 inch Cont Iso Failure
CsI Fract Node 7
1.00E+000
5.55E-017
-1.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

```
C TEST
TITLE
        SHNPP - Early Cont Failure
        END TITLE

ATTACH ATTACH.SAM

PARAMETER CHANGES
        TDMAX   5  SECONDS
        PCF 145. PSI
        ZWSGL 29.7
        ZWCTLB 33. FT
        ZWCTLU 33. FT
        MWSG0 97000. LB
        WFWMX 1.493E5 LB/HR
C
        20,3,0
        2 5.
        END OF PARAMETER CHANGES AND NOLIST

NOT A RESTART

PRINT TIME 6 HOURS

FINAL TIME 24.0

PARALLEL

WHEN BEGIN
     SCRAM ON
     MAIN COOLANT PUMPS OFF
     CHARGING PUMPS OFF
     PZR HEATERS OFF
     PZR SPRAYS OFF
     LPI OFF
     CON SPRAYS OFF
     FAN COOLERS OFF
     MAKEUP OFF
     LETDOWN OFF
     AFW OFF
     END

WHEN TIME > 1.5 HOURS
     BREAK ON
     END

WHEN TIME > 4.0 HOURS
     AFW OFF
     END

INTERVENTION 49
WHEN RPV FAILED IS TRUE
     PARAMETER CHANGES
         ACFPR  1.0 FT**2
         PCF 14.7 PSI
     END
FULL OUTPUT
```

```
REPORT
END

INTERVENTION 50
WHEN CONTMT FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 51
WHEN CORE UNCOVERED IS TRUE
LET CORE UNC TIME = TIME
END

INTERVENTION 52
WHEN HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
  OR HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
LET MELT ONSET TIME = TIME
END

INTERVENTION 53
WHEN PEAK CORE TEMP > TCRHOT
LET PEAK CORE TIME = TIME
RESTORE
SILENT
END

INTERVENTION 54
WHEN PEAK CONTMT PRESSURE = PA
LET PEAK PA TIME = TIME
RESTORE
SILENT
END

INTERVENTION 55
WHEN IEVNT(103) IS TRUE
 OR IEVNT(79) IS TRUE
 PARAMETER CHANGE
  FLPHI  10.0
END
RESTORE 56
END

INTERVENTION 56
WHEN IEVNT(103) IS FALSE
 AND IEVNT(79) IS FALSE
 PARAMETER CHANGE
  FLPHI  2.0
END
RESTORE 55
END

INTERVENTION 57
WHEN IEVNT(42) IS TRUE
 INCREMENT PZR SVS OPEN BY 1
RESTORE 58
END
```

```
INTERVENTION 58
WHEN IEVNT(42) IS FALSE
RESTORE 57
END

WHEN PA > 99.7 PSI
LET CONTMT OVERPRESS TIME = TIME
FULL OUTPUT
REPORT
END
```

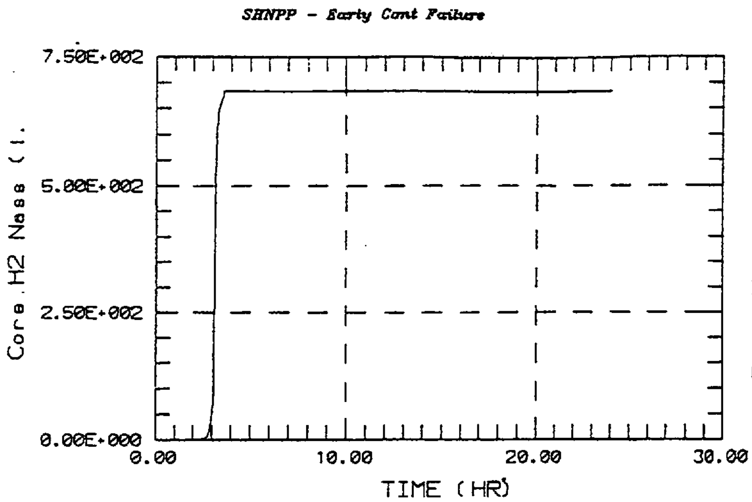
Early Containment Failure

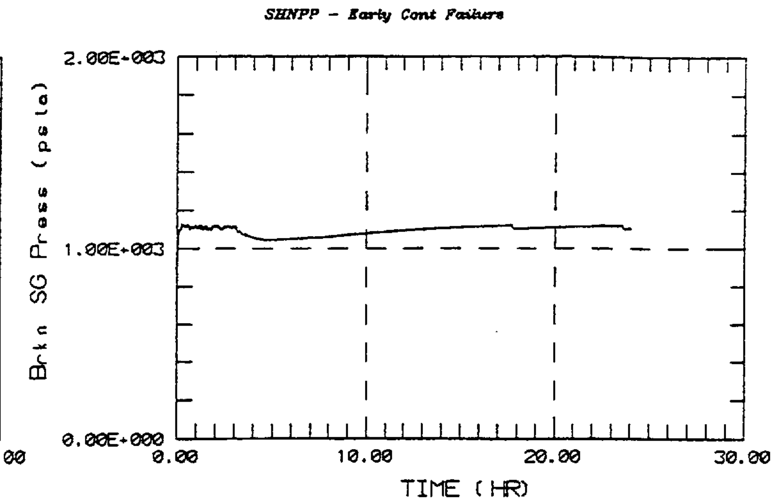
0.0	4	MAIN COOLANT PUMPS OFF
0.0	13	REACTOR SCRAM
0.0	46	LETDOWN FLOW OFF
0.0	156	MSIV CLOSED
0.0	178	AUX CO2 SUPLY DEPLETD
0.0	190	UHI ACCUM EMPTY
0.0	215	MCP SWITCH OFF OR HI-VIBR TRIP
0.0	217	LPI TRAIN 1 FORCED OFF
0.0	221	FANS/COOLERS FORCED OFF
0.0	222	CONTMT SPRAYS FORCED OFF
0.0	223	PZR SPRAYS FORCED OFF
0.0	224	AUX FEED WATER FORCED OFF
0.0	226	1 PZR HTRS FORCED OFF
0.0	227	MANUAL SCRAM
0.0	232	CHARGING PUMPS FORCED OFF
0.0	242	PS MAKEUP OFF
0.0	243	LETDOWN SWITCH OFF
5.2	162	SEC RV OPEN UNBROKEN S/G'S
6.2	152	SEC RV OPEN BROKEN S/G
20.2	152	SEC RV NOT OPEN BROKEN S/G
20.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
21.9	152	SEC RV OPEN BROKEN S/G
21.9	162	SEC RV OPEN UNBROKEN S/G'S
23.5	152	SEC RV NOT OPEN BROKEN S/G
23.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
25.2	152	SEC RV OPEN BROKEN S/G
25.2	162	SEC RV OPEN UNBROKEN S/G'S
26.9	152	SEC RV NOT OPEN BROKEN S/G
26.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
28.5	152	SEC RV OPEN BROKEN S/G
28.5	162	SEC RV OPEN UNBROKEN S/G'S
30.2	152	SEC RV NOT OPEN BROKEN S/G
30.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
44.4	152	SEC RV OPEN BROKEN S/G
44.4	162	SEC RV OPEN UNBROKEN S/G'S
46.1	152	SEC RV NOT OPEN BROKEN S/G
46.1	162	SEC RV NOT OPEN UNBROKEN S/G'S
72.5	162	(20) SEC RV NOT OPEN UNBROKEN S/G'S
74.1	152	(20) SEC RV NOT OPEN BROKEN S/G
103.8	162	(30) SEC RV NOT OPEN UNBROKEN S/G'S
107.4	152	(30) SEC RV NOT OPEN BROKEN S/G
137.3	152	(40) SEC RV NOT OPEN BROKEN S/G
137.3	162	(40) SEC RV NOT OPEN UNBROKEN S/G'S
166.2	162	(50) SEC RV NOT OPEN UNBROKEN S/G'S
167.9	152	(50) SEC RV NOT OPEN BROKEN S/G
192.3	162	(60) SEC RV NOT OPEN UNBROKEN S/G'S
197.7	152	(60) SEC RV NOT OPEN BROKEN S/G
229.3	152	(70) SEC RV NOT OPEN BROKEN S/G
229.3	162	(70) SEC RV NOT OPEN UNBROKEN S/G'S
257.5	162	(80) SEC RV NOT OPEN UNBROKEN S/G'S
268.6	152	(80) SEC RV NOT OPEN BROKEN S/G
290.0	162	(90) SEC RV NOT OPEN UNBROKEN S/G'S
293.5	152	(90) SEC RV NOT OPEN BROKEN S/G
324.7	152	(100) SEC RV NOT OPEN BROKEN S/G
326.5	162	(100) SEC RV NOT OPEN UNBROKEN S/G'S

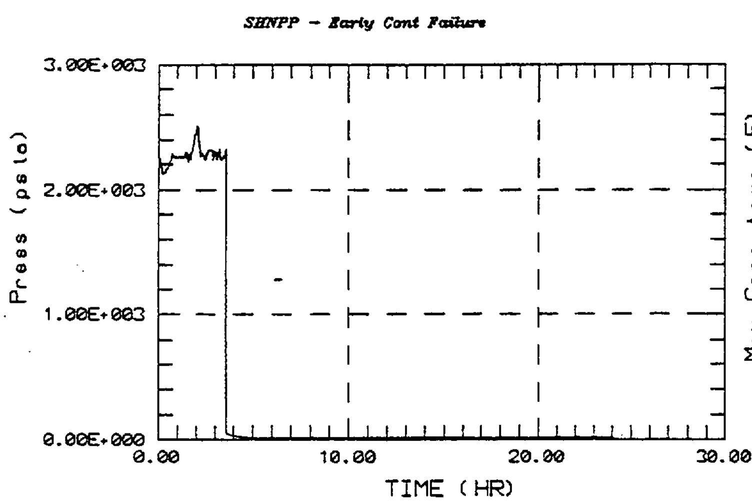
511.9	152	(150)SEC RV NOT OPEN BROKEN S/G
511.9	162	(150)SEC RV NOT OPEN UNBROKEN S/G'S
721.6	152	(200)SEC RV NOT OPEN BROKEN S/G
721.6	162	(200)SEC RV NOT OPEN UNBROKEN S/G'S
953.7	152	(250)SEC RV NOT OPEN BROKEN S/G
953.7	162	(250)SEC RV NOT OPEN UNBROKEN S/G'S
1229.4	152	(300)SEC RV NOT OPEN BROKEN S/G
1229.4	162	(300)SEC RV NOT OPEN UNBROKEN S/G'S
1546.8	152	(350)SEC RV NOT OPEN BROKEN S/G
1546.8	162	(350)SEC RV NOT OPEN UNBROKEN S/G'S
1890.3	152	(400)SEC RV NOT OPEN BROKEN S/G
1890.3	162	(400)SEC RV NOT OPEN UNBROKEN S/G'S
2286.1	162	(450)SEC RV NOT OPEN UNBROKEN S/G'S
2288.4	152	(450)SEC RV NOT OPEN BROKEN S/G
2742.0	152	(500)SEC RV NOT OPEN BROKEN S/G
2742.0	162	(500)SEC RV NOT OPEN UNBROKEN S/G'S
2878.0	44	PZR RELIEF VALVE(S) OPEN
2881.7	44	PZR RELIEF VALVES CLOSED
3131.1	44	PZR RELIEF VALVE(S) OPEN
3134.6	44	PZR RELIEF VALVES CLOSED
3318.3	44	PZR RELIEF VALVE(S) OPEN
3321.6	44	PZR RELIEF VALVES CLOSED
3468.8	44	PZR RELIEF VALVE(S) OPEN
3471.9	44	PZR RELIEF VALVES CLOSED
3604.1	44	PZR RELIEF VALVE(S) OPEN
3607.0	44	PZR RELIEF VALVES CLOSED
4025.9	44	(20)PZR RELIEF VALVES CLOSED
4276.9	44	(30)PZR RELIEF VALVES CLOSED
4443.1	44	(40)PZR RELIEF VALVES CLOSED
4555.4	44	(50)PZR RELIEF VALVES CLOSED
4680.9	44	(60)PZR RELIEF VALVES CLOSED
4957.5	39	PZR EQUIL THERMO
4957.5	40	PZR SOLID
5085.4	44	(70)PZR RELIEF VALVES CLOSED
5341.9	92	Q/T RUPTURE DISK FAILED
5343.5	14	FP MODELS ON
5400.0	209	PS BREAK(S) FAILED
5513.7	44	(80)PZR RELIEF VALVES CLOSED
5552.7	81	WATER ON LOWER CMPT FLOOR
5738.1	151	BROKEN S/G DRY
5739.4	161	UNBKN S/G DRY
5906.0	44	(90)PZR RELIEF VALVES CLOSED
6327.8	44	(100)PZR RELIEF VALVES CLOSED
7082.1	42	PZR SAFETY VALVE(S) OPEN
7167.1	40	PZR HAS STEAM
7200.7	40	PZR SOLID
7229.3	40	PZR HAS STEAM
7266.9	40	PZR SOLID
7291.7	40	PZR HAS STEAM
7330.4	40	PZR SOLID
7355.4	40	PZR HAS STEAM
7395.3	40	PZR SOLID
7411.0	40	PZR HAS STEAM
7438.7	40	PZR SOLID
7659.6	40	(20)PZR SOLID
7749.2	57	WATER IN CAVITY
7825.0	42	PZR SAFETY VALVES CLOSED

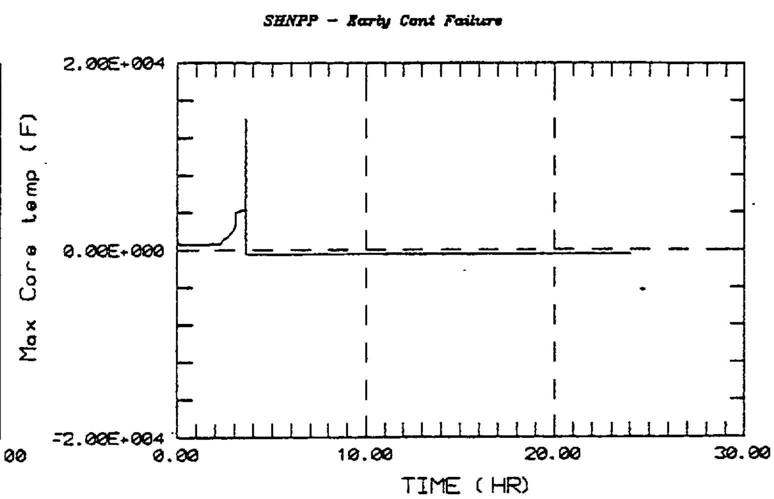
7990.0	25	PS NONEQ THERMO
8020.0	49	CORE HAS UNCOV
8221.6	44	PZR RELIEF VALVES CLOSED
8311.6	44	PZR RELIEF VALVE(S) OPEN
8481.6	44	PZR RELIEF VALVES CLOSED
8571.6	44	PZR RELIEF VALVE(S) OPEN
8706.6	44	PZR RELIEF VALVES CLOSED
8811.6	44	PZR RELIEF VALVE(S) OPEN
8926.6	44	PZR RELIEF VALVES CLOSED
9021.6	152	SEC RV OPEN BROKEN S/G
9023.7	152	SEC RV NOT OPEN BROKEN S/G
9049.8	44	PZR RELIEF VALVE(S) OPEN
9149.8	44	PZR RELIEF VALVES CLOSED
9219.8	162	SEC RV OPEN UNBROKEN S/G'S
9221.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
9293.1	44	PZR RELIEF VALVE(S) OPEN
9323.1	152	SEC RV OPEN BROKEN S/G
9325.1	152	SEC RV NOT OPEN BROKEN S/G
9501.2	162	SEC RV OPEN UNBROKEN S/G'S
9503.2	162	SEC RV NOT OPEN UNBROKEN S/G'S
9674.3	152	SEC RV OPEN BROKEN S/G
9676.3	152	SEC RV NOT OPEN BROKEN S/G
9777.4	162	SEC RV OPEN UNBROKEN S/G'S
9779.4	162	SEC RV NOT OPEN UNBROKEN S/G'S
9965.8	152	SEC RV OPEN BROKEN S/G
9967.8	152	SEC RV NOT OPEN BROKEN S/G
10048.8	162	SEC RV OPEN UNBROKEN S/G'S
10050.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
10111.0	39	PZR NONEQ THERMO
10261.6	152	SEC RV OPEN BROKEN S/G
10263.6	152	SEC RV NOT OPEN BROKEN S/G
10323.9	162	SEC RV OPEN UNBROKEN S/G'S
10325.8	162	SEC RV NOT OPEN UNBROKEN S/G'S
10516.1	44	(20)PZR RELIEF VALVE(S) OPEN
10555.8	32	PZR EMPTY
12848.0	2	SUPPORT PLATE FAILED
12899.3	44	PZR RELIEF VALVE(S) OPEN
12908.0	3	RV FAILED
12908.0	104	CONTMT FAILED
12908.0	61	CORIUM IN CAVITY
12908.2	69	WATER-CORIUM INTERACTION HAS OCCURED IN CAVITY
12908.2	59	WATER FLOODING IN CAVITY TO B
12908.2	59	WATER NOT FLOODING IN CAVITY TO B
12911.2	44	PZR RELIEF VALVES CLOSED
12911.5	58	CORIUM FLOODING IN CAVITY TO B
12911.5	59	WATER FLOODING IN CAVITY TO B
12911.5	58	CORIUM NOT FLOODING IN CAVITY TO B
12911.5	59	WATER NOT FLOODING IN CAVITY TO B
12911.5	82	CORIUM IN LOWER CMPT
12911.5	58	CORIUM FLOODING IN CAVITY TO B
12911.5	59	WATER FLOODING IN CAVITY TO B
12912.4	27	UNBKN LOOPS NOT BLOCKED AT PUMP BOWLS
12913.0	28	DWNCMR NOT BLCKD FOR GAS XPORT
12914.1	5	HPI ON
12915.5	57	CAVITY DRY
12916.0	58	CORIUM NOT FLOODING IN CAVITY TO B
12916.0	61	NO CORIUM IN CAVITY

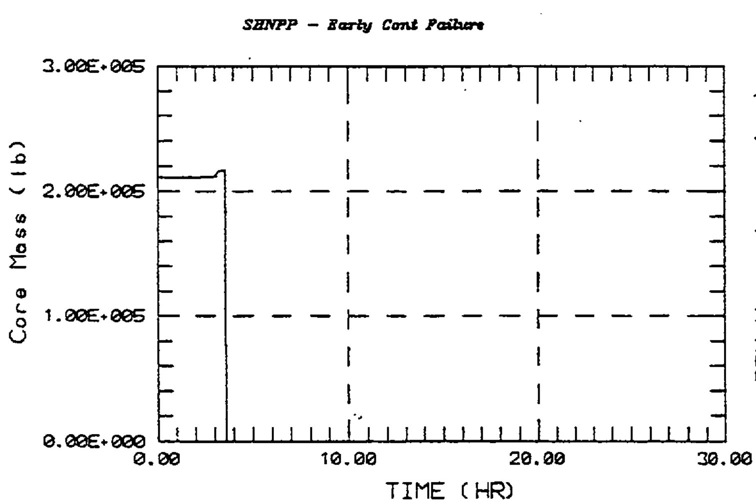
12956.4	61	CORIUM IN CAVITY
13045.1	27	UNBKN LOOPS BLOCKED
39215.8	68	CAVITY SOLID
39215.8	65	CAV CPLD MODEL USED
50592.1	185	HPI PUMPS INSUFF NPSH
50592.1	187	RWST WATER DEPLETED
63740.0	152	SEC RV OPEN BROKEN S/G
63741.9	152	SEC RV NOT OPEN BROKEN S/G
65147.6	162	SEC RV OPEN UNBROKEN S/G'S
65149.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
82565.1	162	SEC RV OPEN UNBROKEN S/G'S
82567.0	162	SEC RV NOT OPEN UNBROKEN S/G'S
84997.6	152	SEC RV OPEN BROKEN S/G
84999.5	152	SEC RV NOT OPEN BROKEN S/G

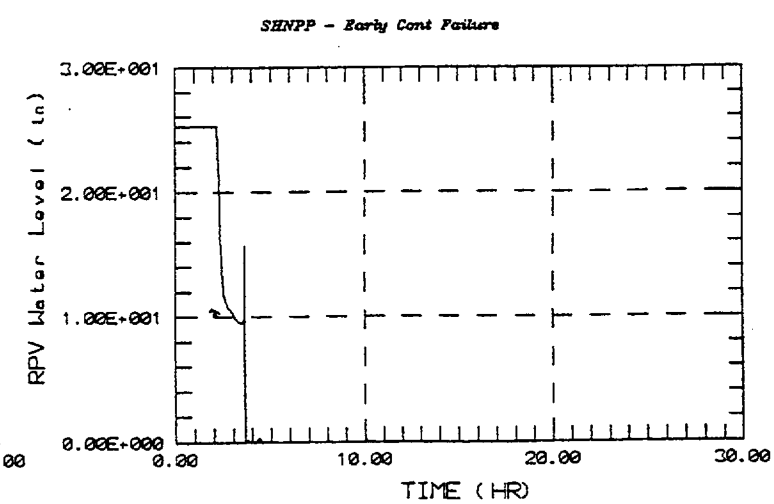

SHNPP - Early Cont Failure
Core H2 Mass (l.
7.50E+002
5.00E+002
2.50E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

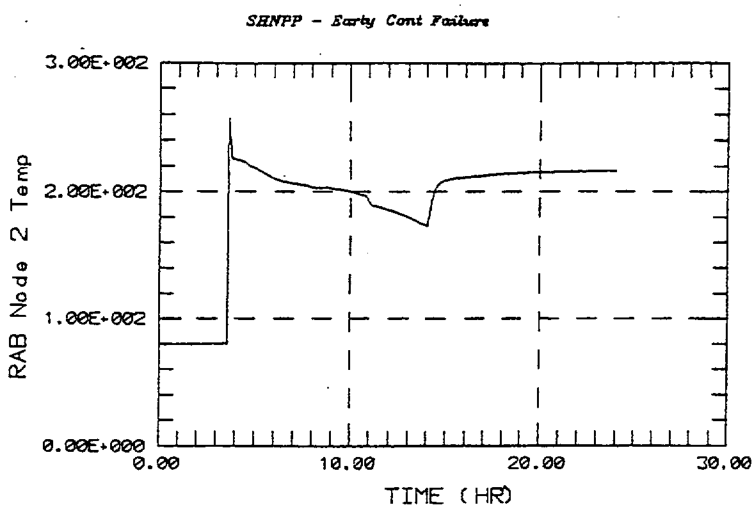

SHNPP - Early Cont Failure
Brkn SG Press (psia)
2.00E+003
1.00E+003
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

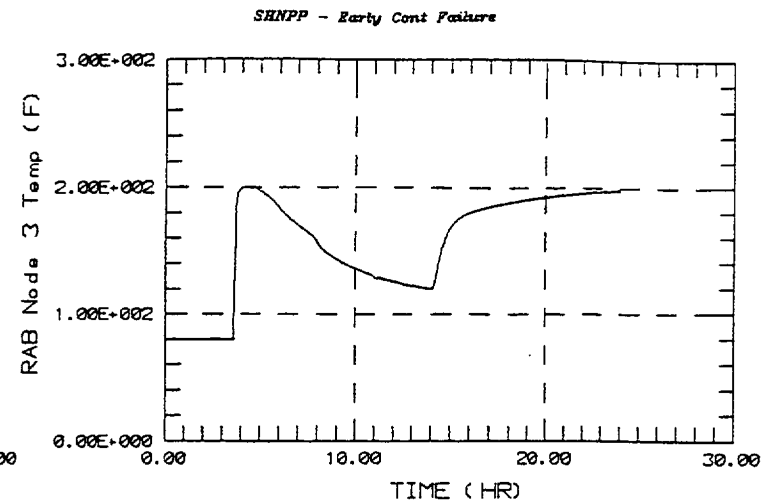

SHNPP - Early Cont Failure
Press (psia)
3.00E+003
2.00E+003
1.00E+003
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

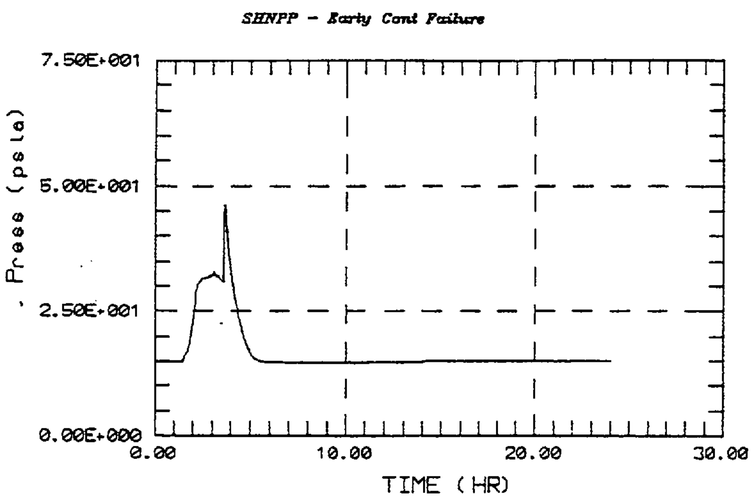

SHNPP - Early Cont Failure
Max Core temp (F)
2.00E+004
0.00E+000
-2.00E+004
0.00
10.00
20.00
30.00
TIME (HR)

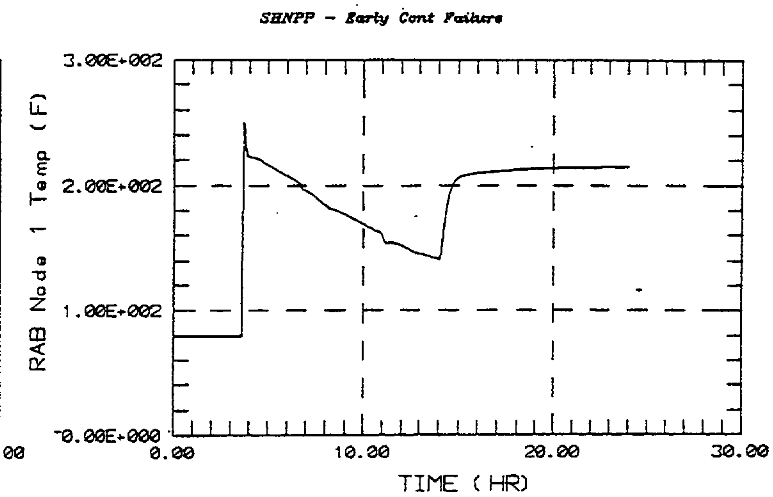

SHNPP - Early Cont Failure
Core Mass (lb)
3.00E+005
2.00E+005
1.00E+005
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

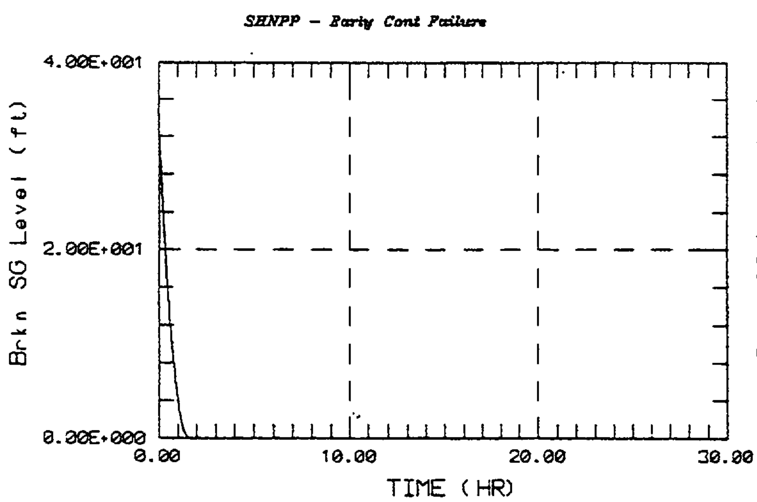

SHNPP - Early Cont Failure
RPV Water Level (in)
3.00E+001
2.00E+001
1.00E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

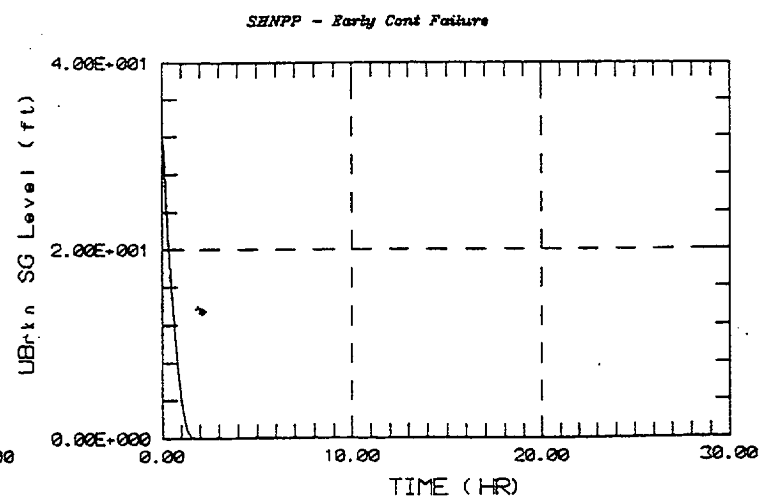

SHNPP - Early Cont Failure
RAB Node 2 Temp
3.00E+002
2.00E+002
1.00E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

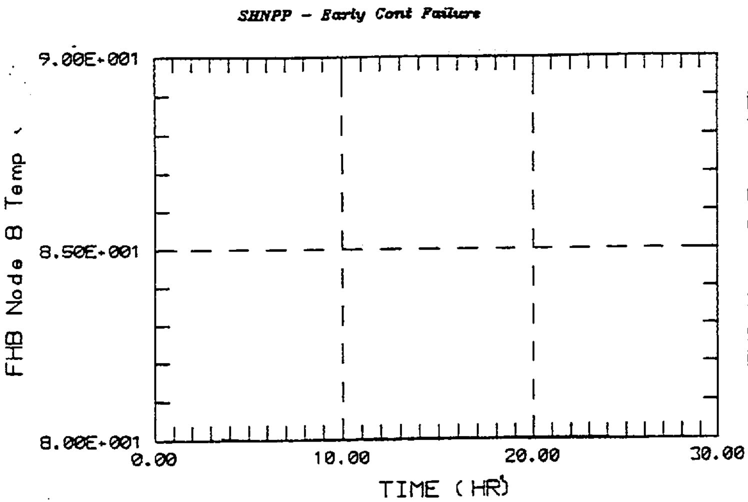

SHNPP - Early Cont Failure
RAB Node 3 Temp (F)
3.00E+002
2.00E+002
1.00E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

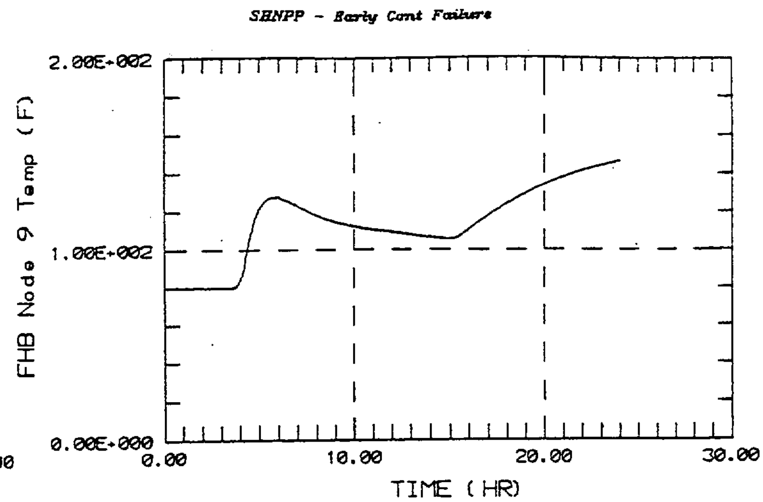

SHNPP - Early Cont Failure
Press (psia)
7.50E+001
5.00E+001
2.50E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

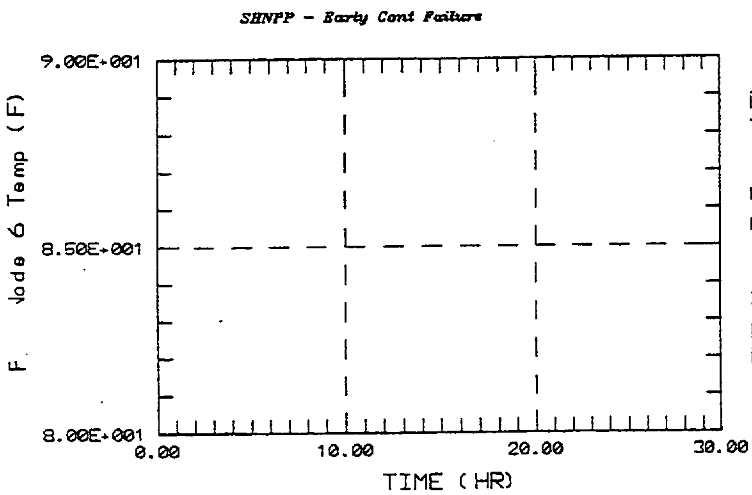

SHNPP - Early Cont Failure
RAB Node 1 Temp (F)
3.00E+002
2.00E+002
1.00E+002
-0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

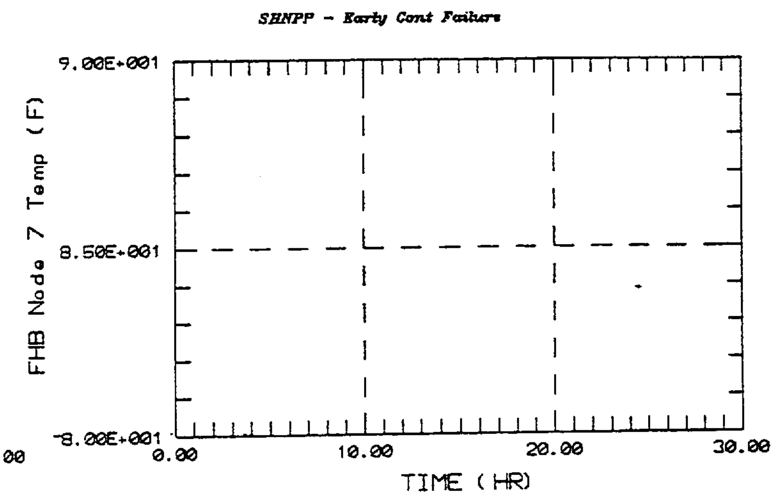

SHNPP - Early Cont Failure
Brkn SG Level (ft)
4.00E+001
2.00E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

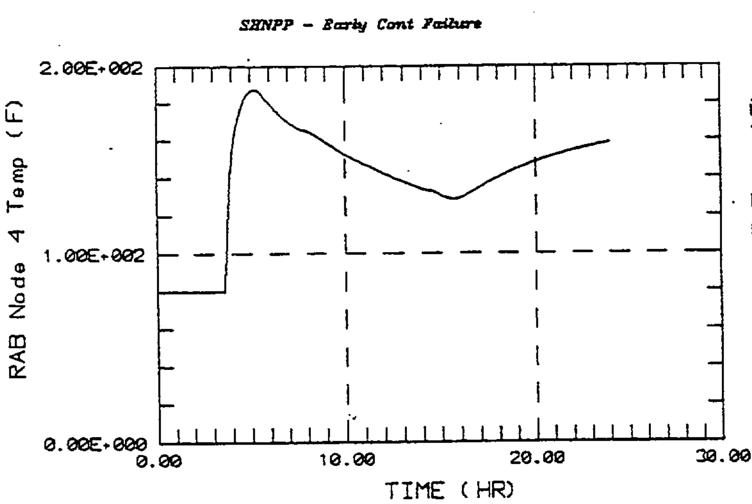

SHNPP - Early Cont Failure
UBrkn SG Level (ft)
4.00E+001
2.00E+001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

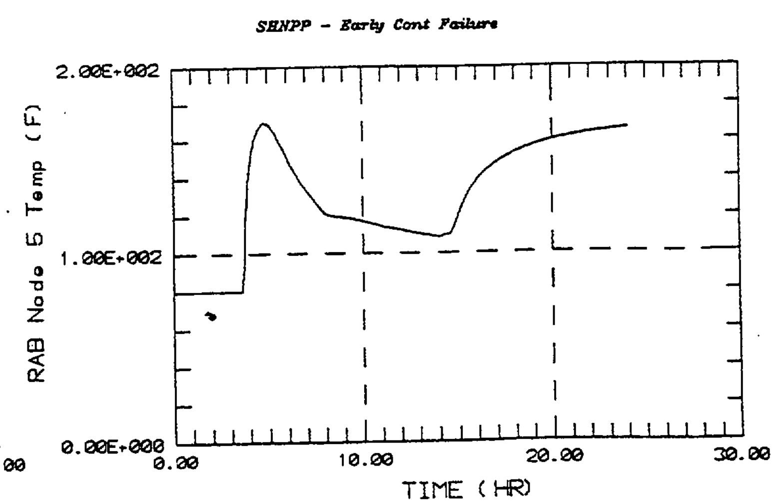

SHNPP - Early Cont Failure
FHB Node 8 Temp
9.00E+001
8.50E+001
8.00E+001
0.00
10.00
20.00
30.00
TIME (HR)

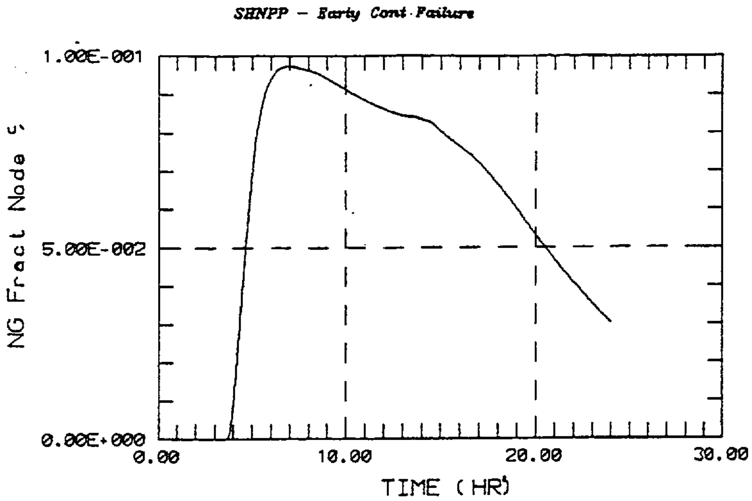

SHNPP - Early Cont Failure
FHB Node 9 Temp (F)
2.00E+002
1.00E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

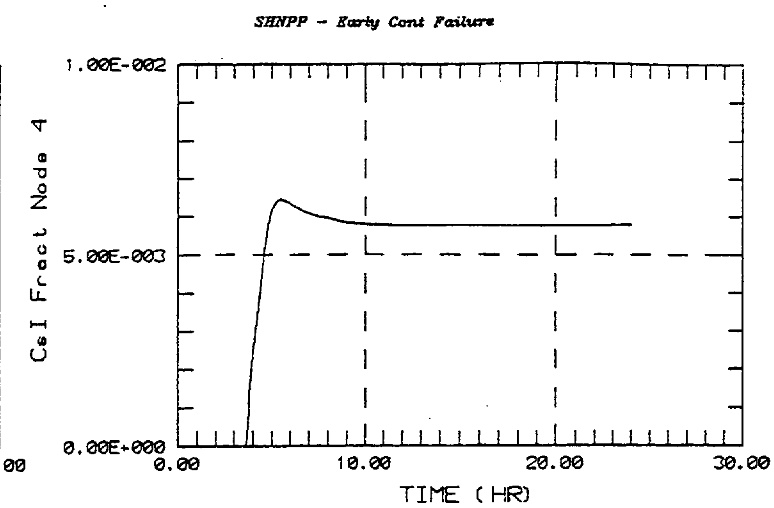

SHNPP - Early Cont Failure
Node 6 Temp (F)
9.00E+001
8.50E+001
8.00E+001
0.00
10.00
20.00
30.00
TIME (HR)

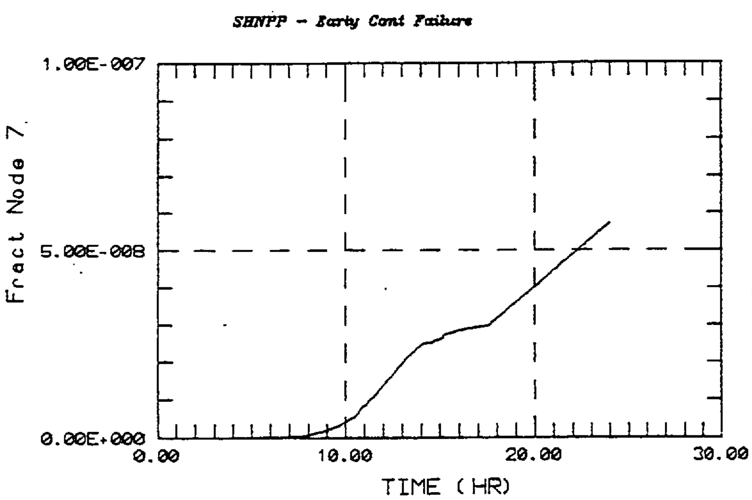

SHNPP - Early Cont Failure
FHB Node 7 Temp (F)
9.00E+001
8.50E+001
8.00E+001
0.00
10.00
20.00
30.00
TIME (HR)

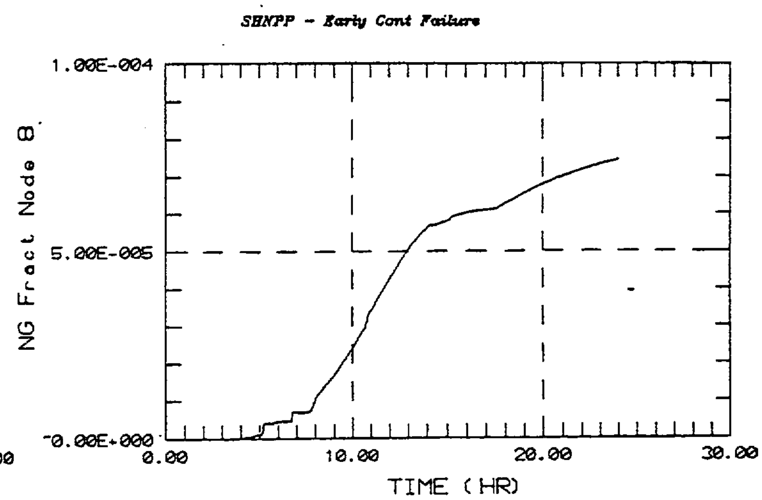

SHNPP - Early Cont Failure
RAB Node 4 Temp (F)
2.00E+002
1.00E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

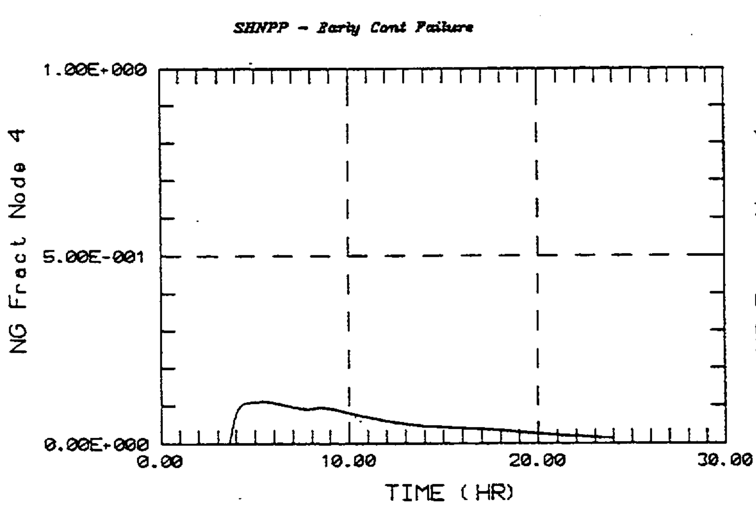

SHNPP - Early Cont Failure
RAB Node 5 Temp (F)
2.00E+002
1.00E+002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

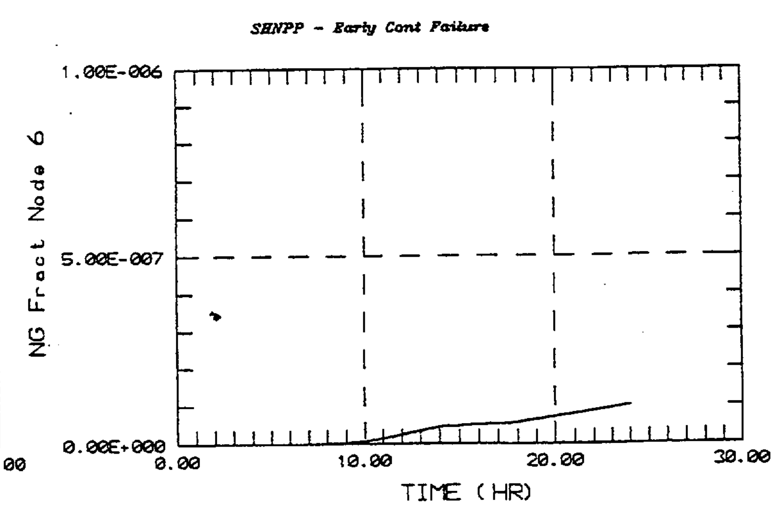

SHNPP - Early Cont Failure
NG Fract Node 5
1.00E-001
5.00E-002
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

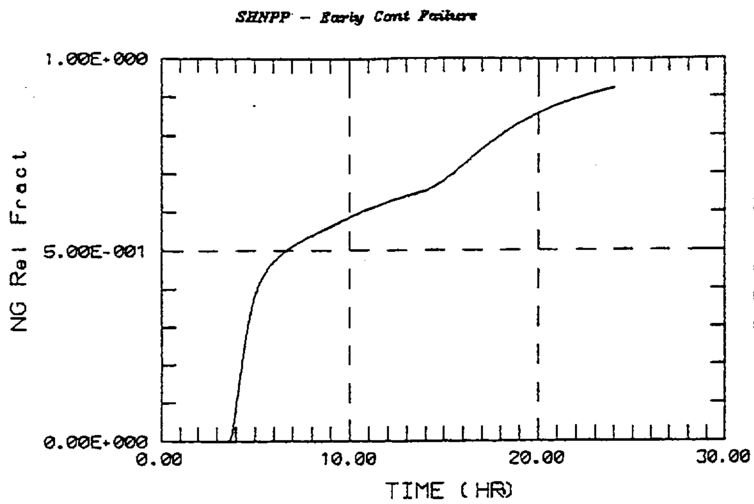

SHNPP - Early Cont Failure
CsI Fract Node 4
1.00E-002
5.00E-003
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

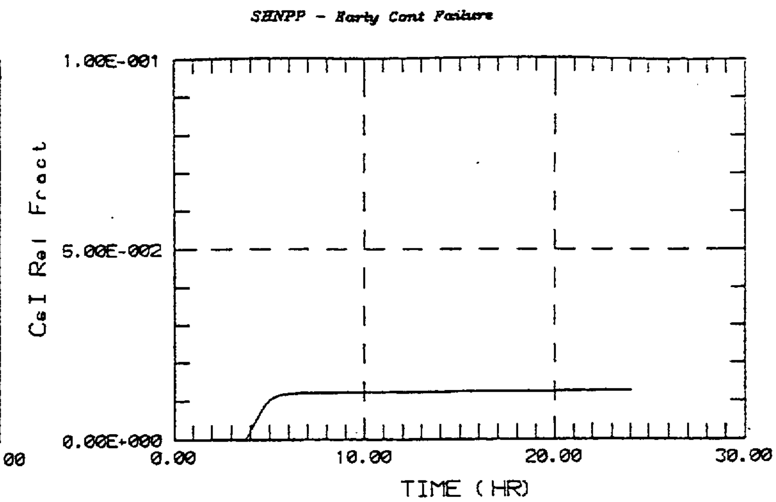

SHNPP - Early Cont Failure
Fract Node 7
1.00E-007
5.00E-008
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

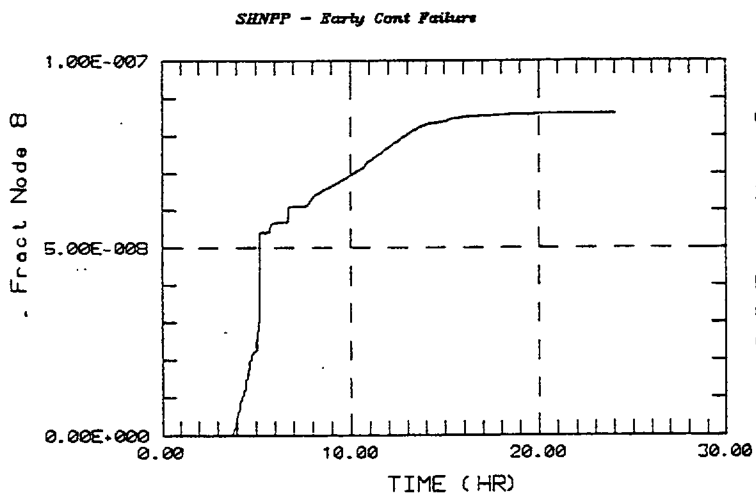

SHNPP - Early Cont Failure
NG Fract Node 8
1.00E-004
5.00E-005
-0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

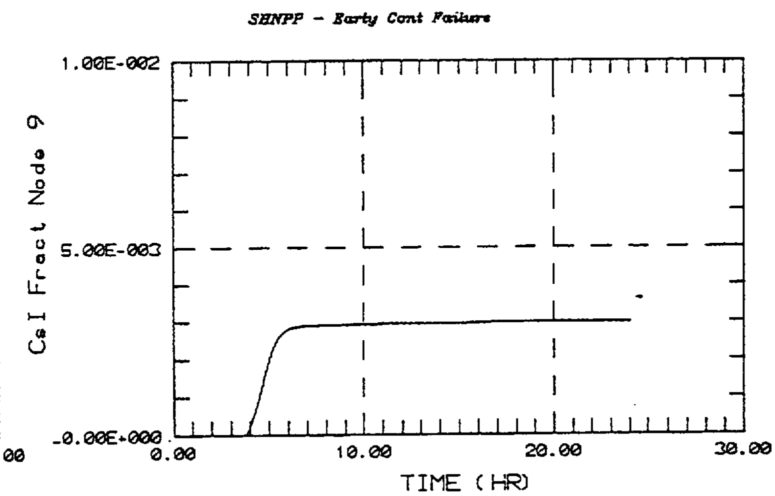

SHNPP - Early Cont Failure
NG Fract Node 4
1.00E+000
5.00E-001
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

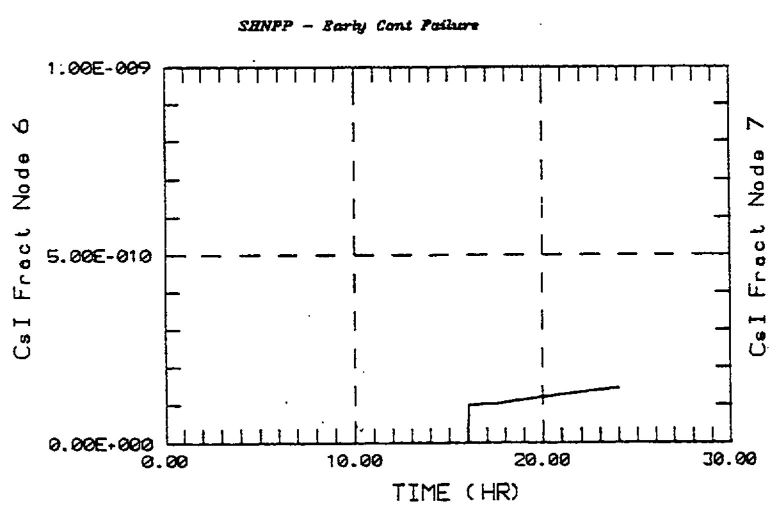

SHNPP - Early Cont Failure
NG Fract Node 6
1.00E-006
5.00E-007
0.00E+000
0.00
10.00
20.00
30.00
TIME (HR)

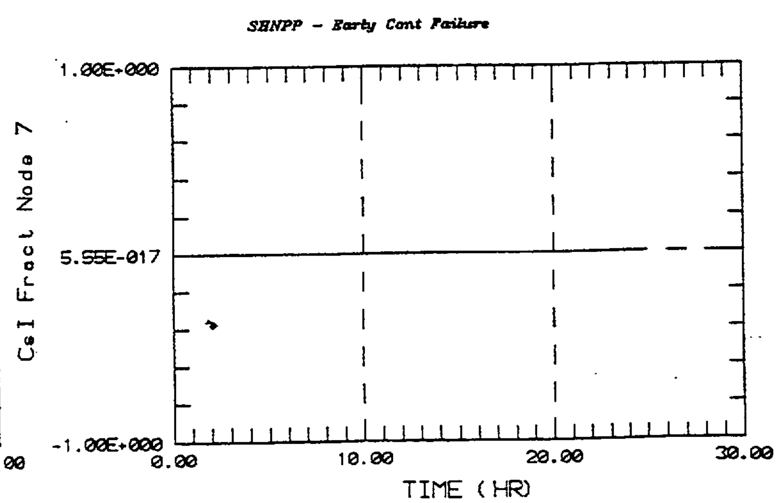

SHNPP - Early Cont Failure
1.00E+000
5.00E-001
0.00E+000
NG Rel Fract
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - Early Cont Failure
1.00E-001
5.00E-002
0.00E+000
CsI Rel Fract
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - Early Cont Failure
1.00E-007
5.00E-008
0.00E+000
Fract Node 8
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - Early Cont Failure
1.00E-002
5.00E-003
-0.00E+000
CsI Fract Node 9
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - Early Cont Failure
1.00E-009
5.00E-010
0.00E+000
CsI Fract Node 6
0.00
10.00
20.00
30.00
TIME (HR)


SHNPP - Early Cont Failure
1.00E+000
5.55E-017
-1.00E+000
CsI Fract Node 7
0.00
10.00
20.00
30.00
TIME (HR)

```
C TEST
TITLE
        SHNPP - Late Cont Failure
        END TITLE

ATTACH ATTACH.SAM

PARAMETER CHANGES
        TDMAX    5  SECONDS
        PCF 145. PSI
        BREAK NODE 7
        BREAK AREA 0.04909 FT**2
        BREAK ELEVATION 26.4 FT
        20,3,0
        2 5.
        END OF PARAMETER CHANGES AND NOLIST

NOT A RESTART

PRINT TIME 10 HOURS

FINAL TIME 60.0

PARALLEL

WHEN BEGIN
     SCRAM ON
     MAIN COOLANT PUMPS OFF
     BREAK ON
     PZR HEATERS OFF
     PZR SPRAYS OFF
     MAIN FEEDWATER PUMPS OFF
     CON SPRAYS OFF
     FAN COOLERS OFF
     LPI OFF
     END

WHEN ZWRWST < 9.31 FT
     LPI OFF
     HPI OFF
     RECIRCULATION MODE ON
     END

WHEN TCRHOT > 2100. f
    AND TIME > 10. MIN
    REPORT
     END

INTERVENTION 47
WHEN SCRAM IS TRUE
 PARAMETER CHANGE
  ZWCTLB 41. FT
  ZWCTLU 41. FT
  END
END
```

```
INTERVENTION 49
WHEN RPV FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 50
WHEN CONTMT FAILED IS TRUE
FULL OUTPUT
REPORT
END

INTERVENTION 51
WHEN CORE UNCOVERED IS TRUE
LET CORE UNC TIME = TIME
END

INTERVENTION 52
WHEN HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
  OR HOTTEST CORE TEMP > %EUTECTIC TEMPERATURE%
LET MELT ONSET TIME = TIME
END

INTERVENTION 53
WHEN PEAK CORE TEMP > TCRHOT
LET PEAK CORE TIME = TIME
RESTORE
SILENT
END

INTERVENTION 54
WHEN PEAK CONTMT PRESSURE = PA
LET PEAK PA TIME = TIME
RESTORE
SILENT
END

INTERVENTION 55
WHEN IEVNT(103) IS TRUE
 OR IEVNT(79) IS TRUE
 PARAMETER CHANGE
  FLPHI  10.0
END
RESTORE 56
END

INTERVENTION 56
WHEN IEVNT(103) IS FALSE
 AND IEVNT(79) IS FALSE
 PARAMETER CHANGE
  FLPHI  2.0
END
RESTORE 55
END
```

```
INTERVENTION 57
WHEN IEVNT(42) IS TRUE
 INCREMENT PZR SVS OPEN BY 1
RESTORE 58
END

INTERVENTION 58
WHEN IEVNT(42) IS FALSE
RESTORE 57
END

WHEN PA > 99.7 PSI
LET CONTMT OVERPRESS TIME = TIME
FULL OUTPUT
REPORT
END
```

Late Containment Failure

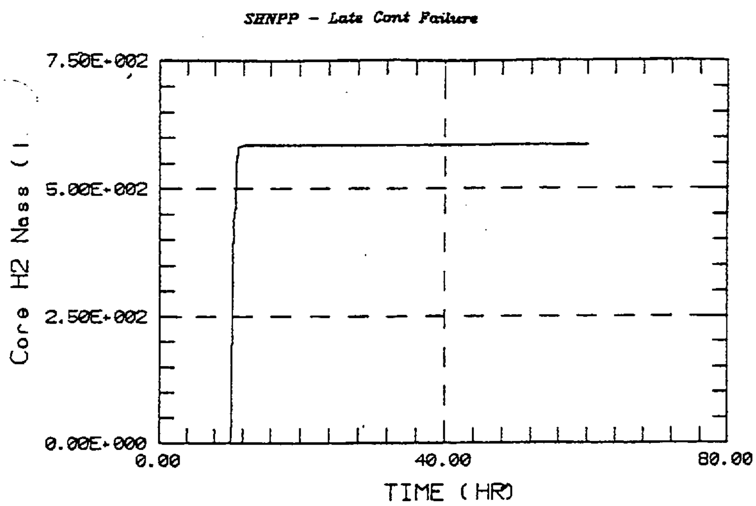
0.0	4	MAIN COOLANT PUMPS OFF
0.0	13	REACTOR SCRAM
0.0	154	AUX FEEDWATER ON
0.0	156	MSIV CLOSED
0.0	178	AUX CO2 SUPLY DEPLETD
0.0	190	UHI ACCUM EMPTY
0.0	209	PS BREAK(S) FAILED
0.0	215	MCP SWITCH OFF OR HI-VIBR TRIP
0.0	217	LPI TRAIN 1 FORCED OFF
0.0	221	FANS/COOLERS FORCED OFF
0.0	222	CONTMT SPRAYS FORCED OFF
0.0	223	PZR SPRAYS FORCED OFF
0.0	226	1 PZR HTRS FORCED OFF
0.0	227	MANUAL SCRAM
0.0	228	MAIN FW SHUT OFF
0.3	14	FP MODELS ON
0.5	81	WATER ON LOWER CMPT FLOOR
5.6	162	SEC RV OPEN UNBROKEN S/G'S
7.4	152	SEC RV OPEN BROKEN S/G
9.0	162	SEC RV NOT OPEN UNBROKEN S/G'S
9.6	152	SEC RV NOT OPEN BROKEN S/G
9.6	162	SEC RV OPEN UNBROKEN S/G'S
10.3	152	SEC RV OPEN BROKEN S/G
10.3	162	SEC RV NOT OPEN UNBROKEN S/G'S
11.1	152	SEC RV NOT OPEN BROKEN S/G
11.1	162	SEC RV OPEN UNBROKEN S/G'S
11.7	162	SEC RV NOT OPEN UNBROKEN S/G'S
21.9	152	SEC RV OPEN BROKEN S/G
22.7	152	SEC RV NOT OPEN BROKEN S/G
23.6	162	SEC RV OPEN UNBROKEN S/G'S
24.4	162	SEC RV NOT OPEN UNBROKEN S/G'S
25.2	152	SEC RV OPEN BROKEN S/G
26.0	152	SEC RV NOT OPEN BROKEN S/G
26.7	162	SEC RV OPEN UNBROKEN S/G'S
27.5	162	SEC RV NOT OPEN UNBROKEN S/G'S
28.2	152	SEC RV OPEN BROKEN S/G
29.0	152	SEC RV NOT OPEN BROKEN S/G
31.1	5	HPI ON
39.0	162	(20) SEC RV NOT OPEN UNBROKEN S/G'S
39.5	152	(20) SEC RV NOT OPEN BROKEN S/G
49.2	152	(30) SEC RV NOT OPEN BROKEN S/G
49.2	162	(30) SEC RV NOT OPEN UNBROKEN S/G'S
59.4	152	(40) SEC RV NOT OPEN BROKEN S/G
59.4	162	(40) SEC RV NOT OPEN UNBROKEN S/G'S
69.9	32	PZR EMPTY
75.5	32	PZR NOT EMPTY
77.1	32	PZR EMPTY
83.0	32	PZR NOT EMPTY
89.8	152	(50) SEC RV NOT OPEN BROKEN S/G
89.8	162	(50) SEC RV NOT OPEN UNBROKEN S/G'S
93.4	32	PZR EMPTY
173.3	152	(60) SEC RV NOT OPEN BROKEN S/G
173.3	162	(60) SEC RV NOT OPEN UNBROKEN S/G'S
740.0	57	WATER IN CAVITY
994.4	152	(70) SEC RV NOT OPEN BROKEN S/G

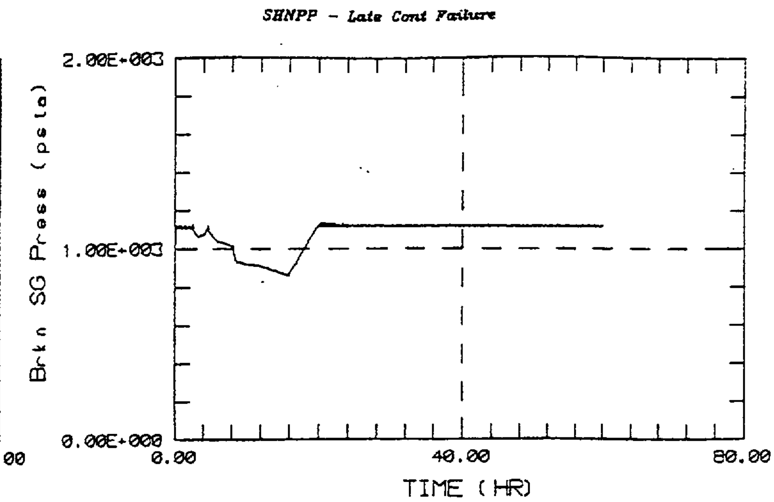
1070.7	162	(70)SEC RV NOT OPEN UNBROKEN S/G'S
2030.6	152	(80)SEC RV NOT OPEN BROKEN S/G
2106.2	162	(80)SEC RV NOT OPEN UNBROKEN S/G'S
2857.8	32	PZR NOT EMPTY
3284.0	152	(90)SEC RV NOT OPEN BROKEN S/G
3897.7	162	SEC RV OPEN UNBROKEN S/G'S
3898.9	162	SEC RV NOT OPEN UNBROKEN S/G'S
5251.4	152	SEC RV OPEN BROKEN S/G
5252.6	152	SEC RV NOT OPEN BROKEN S/G
6081.1	162	SEC RV OPEN UNBROKEN S/G'S
6082.3	162	SEC RV NOT OPEN UNBROKEN S/G'S
7680.6	152	SEC RV OPEN BROKEN S/G
7681.8	152	SEC RV NOT OPEN BROKEN S/G
9020.0	152	SEC RV OPEN BROKEN S/G
9020.0	162	SEC RV OPEN UNBROKEN S/G'S
9021.1	152	SEC RV NOT OPEN BROKEN S/G
9021.1	162	SEC RV NOT OPEN UNBROKEN S/G'S
9022.1	152	SEC RV OPEN BROKEN S/G
9022.1	162	SEC RV OPEN UNBROKEN S/G'S
9023.3	152	SEC RV NOT OPEN BROKEN S/G
9023.3	162	SEC RV NOT OPEN UNBROKEN S/G'S
9024.4	152	SEC RV OPEN BROKEN S/G
9024.4	162	SEC RV OPEN UNBROKEN S/G'S
9025.6	152	SEC RV NOT OPEN BROKEN S/G
9025.6	162	SEC RV NOT OPEN UNBROKEN S/G'S
9033.6	39	PZR EQUIL THERMO
11051.5	39	PZR NONEQ THERMO
16548.1	152	SEC RV OPEN BROKEN S/G
16549.2	152	SEC RV NOT OPEN BROKEN S/G
16550.4	152	SEC RV OPEN BROKEN S/G
16551.5	152	SEC RV NOT OPEN BROKEN S/G
16551.5	162	SEC RV OPEN UNBROKEN S/G'S
16552.7	162	SEC RV NOT OPEN UNBROKEN S/G'S
16553.9	162	SEC RV OPEN UNBROKEN S/G'S
16555.0	162	SEC RV NOT OPEN UNBROKEN S/G'S
16558.0	39	PZR EQUIL THERMO
18526.5	65	CAV CPLD MODEL USED
21048.1	39	PZR NONEQ THERMO
29244.7	5	HPI OFF
29244.7	181	RECIRC SYSTEM IN OPERATION
29244.7	216	HPI FORCED OFF
29244.7	220	RECIRC SWITCH: MAN ON
29384.7	39	PZR EQUIL THERMO
29396.6	39	PZR NONEQ THERMO
29747.5	32	PZR EMPTY
30842.5	188	ACCUMULATOR WATER DEPLETED
30854.1	32	PZR NOT EMPTY
30863.9	32	PZR EMPTY
30870.4	68	CAVITY SOLID
30883.8	32	PZR NOT EMPTY
30900.0	32	PZR EMPTY
30907.2	32	PZR NOT EMPTY
30916.7	32	PZR EMPTY
30920.0	32	PZR NOT EMPTY
30933.1	32	PZR EMPTY
30945.1	32	PZR NOT EMPTY
30948.3	32	PZR EMPTY

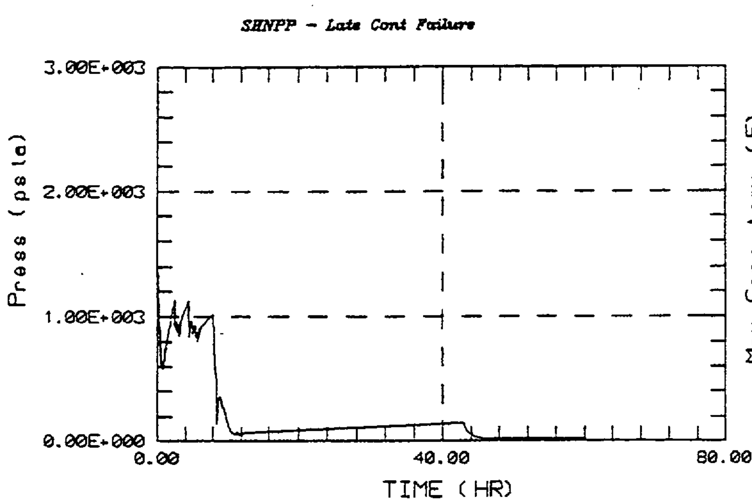
30959.1	32	PZR NOT EMPTY
30982.3	32	PZR EMPTY
30992.8	32	PZR NOT EMPTY
31004.4	32	PZR EMPTY
31017.1	32	PZR NOT EMPTY
31075.8	32	PZR EMPTY
31081.4	32	PZR NOT EMPTY
32967.4	25	PS NONEQ THERMO
33951.2	32	PZR EMPTY
34656.2	49	CORE HAS UNCOV
43230.2	2	SUPPORT PLATE FAILED
43290.2	3	RV FAILED
43290.4	61	CORIUM IN CAVITY
43290.8	69	WATER-CORIUM INTERACTION HAS OCCURED IN CAVITY
43290.9	59	WATER FLOODING IN CAVITY TO B
43290.9	59	WATER NOT FLOODING IN CAVITY TO B
43290.9	68	CAVITY NOT FULL
43290.9	68	CAVITY SOLID
43295.2	59	WATER FLOODING IN CAVITY TO B
43295.2	68	CAVITY NOT FULL
43295.2	59	WATER NOT FLOODING IN CAVITY TO B
43295.2	65	CAV UNCPLD MODEL USED
43295.3	68	CAVITY SOLID
43295.3	65	CAV CPLD MODEL USED
43295.3	59	WATER FLOODING IN CAVITY TO B
43295.3	68	CAVITY NOT FULL
43295.5	65	CAV UNCPLD MODEL USED
43296.5	59	WATER NOT FLOODING IN CAVITY TO B
43296.5	59	WATER FLOODING IN CAVITY TO B
43296.6	59	WATER NOT FLOODING IN CAVITY TO B
43296.6	59	WATER FLOODING IN CAVITY TO B
43296.7	59	WATER NOT FLOODING IN CAVITY TO B
43296.7	59	WATER FLOODING IN CAVITY TO B
43296.7	59	WATER NOT FLOODING IN CAVITY TO B
43296.8	59	WATER FLOODING IN CAVITY TO B
43296.8	59	WATER NOT FLOODING IN CAVITY TO B
43296.9	59	WATER FLOODING IN CAVITY TO B
43296.9	59	WATER NOT FLOODING IN CAVITY TO B
43296.9	59	WATER FLOODING IN CAVITY TO B
43297.0	59	WATER NOT FLOODING IN CAVITY TO B
43297.0	59	WATER FLOODING IN CAVITY TO B
43297.1	59	WATER NOT FLOODING IN CAVITY TO B
43297.1	59	WATER FLOODING IN CAVITY TO B
43297.1	59	WATER NOT FLOODING IN CAVITY TO B
43297.2	59	WATER FLOODING IN CAVITY TO B
43297.2	59	WATER NOT FLOODING IN CAVITY TO B
43297.2	59	WATER FLOODING IN CAVITY TO B
43297.3	59	WATER NOT FLOODING IN CAVITY TO B
43297.3	59	WATER FLOODING IN CAVITY TO B
43297.4	59	WATER NOT FLOODING IN CAVITY TO B
43297.4	59	WATER FLOODING IN CAVITY TO B
43297.4	59	WATER NOT FLOODING IN CAVITY TO B
43297.5	59	WATER FLOODING IN CAVITY TO B
43297.5	59	WATER NOT FLOODING IN CAVITY TO B
43297.5	59	WATER FLOODING IN CAVITY TO B
43297.6	59	WATER NOT FLOODING IN CAVITY TO B
43297.6	59	WATER FLOODING IN CAVITY TO B

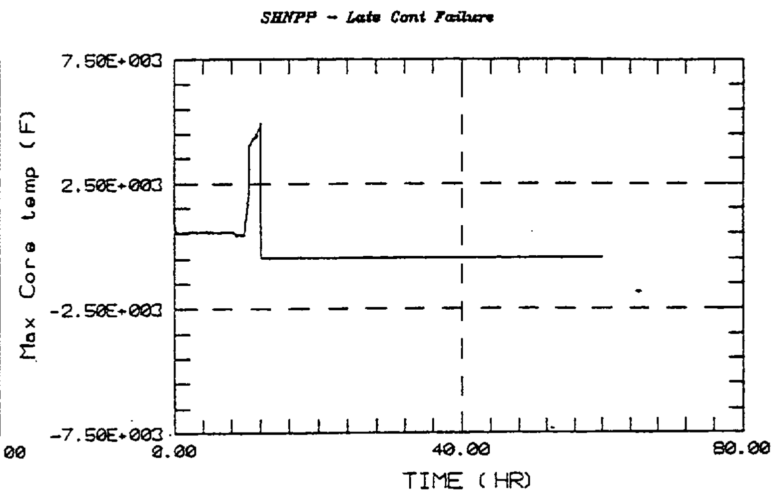
43313.1	59	WATER NOT FLOODING IN CAVITY TO B
43313.1	59	WATER FLOODING IN CAVITY TO B
43313.1	59	WATER NOT FLOODING IN CAVITY TO B
43313.1	59	WATER FLOODING IN CAVITY TO B
43313.2	59	WATER NOT FLOODING IN CAVITY TO B
43313.2	59	WATER FLOODING IN CAVITY TO B
43313.2	59	WATER NOT FLOODING IN CAVITY TO B
43313.2	59	WATER FLOODING IN CAVITY TO B
43313.3	59	WATER NOT FLOODING IN CAVITY TO B
43313.3	59	WATER FLOODING IN CAVITY TO B
43313.3	59	WATER NOT FLOODING IN CAVITY TO B
43313.3	59	WATER FLOODING IN CAVITY TO B
43313.4	59	WATER NOT FLOODING IN CAVITY TO B
43313.4	59	WATER FLOODING IN CAVITY TO B
43313.4	59	WATER NOT FLOODING IN CAVITY TO B
43313.4	59	WATER FLOODING IN CAVITY TO B
43313.5	59	WATER NOT FLOODING IN CAVITY TO B
43313.5	59	WATER FLOODING IN CAVITY TO B
43313.5	59	WATER NOT FLOODING IN CAVITY TO B
43313.5	59	WATER FLOODING IN CAVITY TO B
43313.6	59	WATER NOT FLOODING IN CAVITY TO B
43313.6	59	WATER FLOODING IN CAVITY TO B
43313.6	59	WATER NOT FLOODING IN CAVITY TO B
43313.6	59	WATER FLOODING IN CAVITY TO B
43313.7	59	WATER NOT FLOODING IN CAVITY TO B
43313.7	59	WATER FLOODING IN CAVITY TO B
43313.7	59	WATER NOT FLOODING IN CAVITY TO B
43313.7	59	WATER FLOODING IN CAVITY TO B
43313.8	59	WATER NOT FLOODING IN CAVITY TO B
43313.8	59	WATER FLOODING IN CAVITY TO B
43313.8	59	WATER NOT FLOODING IN CAVITY TO B
43317.9	28	DWNCMR NOT BLCKD FOR GAS XPORT
43319.0	68	CAVITY SOLID
43319.0	65	CAV CPLD MODEL USED
71465.0	152	SEC RV OPEN BROKEN S/G
71466.1	152	SEC RV NOT OPEN BROKEN S/G
71467.2	152	SEC RV OPEN BROKEN S/G
71468.2	152	SEC RV NOT OPEN BROKEN S/G
71469.8	152	SEC RV OPEN BROKEN S/G
71470.8	152	SEC RV NOT OPEN BROKEN S/G
72022.3	152	SEC RV OPEN BROKEN S/G
72023.4	152	SEC RV NOT OPEN BROKEN S/G
72024.4	152	SEC RV OPEN BROKEN S/G
72025.5	152	SEC RV NOT OPEN BROKEN S/G
73103.0	152	(20)SEC RV NOT OPEN BROKEN S/G
73640.1	152	(30)SEC RV NOT OPEN BROKEN S/G
74692.3	152	(40)SEC RV NOT OPEN BROKEN S/G
75740.1	152	(50)SEC RV NOT OPEN BROKEN S/G
76272.5	152	(60)SEC RV NOT OPEN BROKEN S/G
77330.2	152	(70)SEC RV NOT OPEN BROKEN S/G
78229.0	152	(80)SEC RV NOT OPEN BROKEN S/G
79128.0	152	(90)SEC RV NOT OPEN BROKEN S/G
80027.2	152	(100)SEC RV NOT OPEN BROKEN S/G
84650.0	152	(150)SEC RV NOT OPEN BROKEN S/G
89703.6	152	(200)SEC RV NOT OPEN BROKEN S/G
94608.9	152	(250)SEC RV NOT OPEN BROKEN S/G
100169.8	152	(300)SEC RV NOT OPEN BROKEN S/G

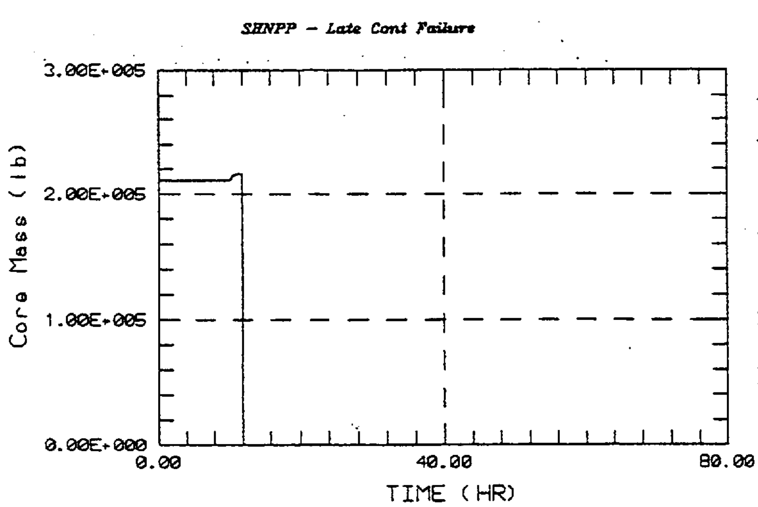
105691.5	152	(350)SEC RV NOT OPEN BROKEN S/G
114812.0	152	(400)SEC RV NOT OPEN BROKEN S/G
123575.0	152	(450)SEC RV NOT OPEN BROKEN S/G
133440.5	152	(500)SEC RV NOT OPEN BROKEN S/G
154592.8	104	CONTMT FAILED
161958.9	68	CAVITY NOT FULL
198816.2	152	SEC RV OPEN BROKEN S/G
198817.4	152	SEC RV NOT OPEN BROKEN S/G
198818.7	152	SEC RV OPEN BROKEN S/G
198820.0	152	SEC RV NOT OPEN BROKEN S/G
199423.7	152	SEC RV OPEN BROKEN S/G
199425.0	152	SEC RV NOT OPEN BROKEN S/G
199426.2	152	SEC RV OPEN BROKEN S/G
199427.4	152	SEC RV NOT OPEN BROKEN S/G
199429.1	152	SEC RV OPEN BROKEN S/G
199430.4	152	SEC RV NOT OPEN BROKEN S/G
200434.0	152	SEC RV OPEN BROKEN S/G
200435.3	152	SEC RV NOT OPEN BROKEN S/G
200436.6	152	SEC RV OPEN BROKEN S/G
200437.8	152	SEC RV NOT OPEN BROKEN S/G
201046.6	152	SEC RV OPEN BROKEN S/G
201047.8	152	SEC RV NOT OPEN BROKEN S/G
201049.1	152	SEC RV OPEN BROKEN S/G
201050.3	152	SEC RV NOT OPEN BROKEN S/G
201052.0	152	SEC RV OPEN BROKEN S/G
201053.3	152	SEC RV NOT OPEN BROKEN S/G
202061.9	152	SEC RV OPEN BROKEN S/G
202063.2	152	SEC RV NOT OPEN BROKEN S/G
202064.5	152	SEC RV OPEN BROKEN S/G
202065.7	152	SEC RV NOT OPEN BROKEN S/G
202674.5	152	SEC RV OPEN BROKEN S/G
202675.7	152	SEC RV NOT OPEN BROKEN S/G
202676.9	152	SEC RV OPEN BROKEN S/G
202678.2	152	SEC RV NOT OPEN BROKEN S/G
202679.9	152	SEC RV OPEN BROKEN S/G
202681.1	152	SEC RV NOT OPEN BROKEN S/G
203694.8	152	SEC RV OPEN BROKEN S/G
203696.1	152	SEC RV NOT OPEN BROKEN S/G
203697.3	152	SEC RV OPEN BROKEN S/G
203698.6	152	SEC RV NOT OPEN BROKEN S/G
204307.3	152	SEC RV OPEN BROKEN S/G
204308.6	152	SEC RV NOT OPEN BROKEN S/G
204309.8	152	SEC RV OPEN BROKEN S/G
204311.0	152	SEC RV NOT OPEN BROKEN S/G
204312.8	152	SEC RV OPEN BROKEN S/G
204314.0	152	SEC RV NOT OPEN BROKEN S/G
205332.7	152	SEC RV OPEN BROKEN S/G
205333.9	152	SEC RV NOT OPEN BROKEN S/G
205335.2	152	SEC RV OPEN BROKEN S/G
205336.5	152	SEC RV NOT OPEN BROKEN S/G
205945.2	152	SEC RV OPEN BROKEN S/G
205946.5	152	SEC RV NOT OPEN BROKEN S/G
205947.7	152	SEC RV OPEN BROKEN S/G
205948.9	152	SEC RV NOT OPEN BROKEN S/G
205950.6	152	SEC RV OPEN BROKEN S/G
205951.9	152	SEC RV NOT OPEN BROKEN S/G
206975.5	152	SEC RV OPEN BROKEN S/G


SHNPP - Late Cont Failure
Core H2 Nass (l
7.50E+002
5.00E+002
2.50E+002
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
Brkn SG Press (psia)
2.00E+003
1.00E+003
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
Press (psia)
3.00E+003
2.00E+003
1.00E+003
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
Max Core temp (F)
7.50E+003
2.50E+003
-2.50E+003
-7.50E+003
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
Core Mass (lb)
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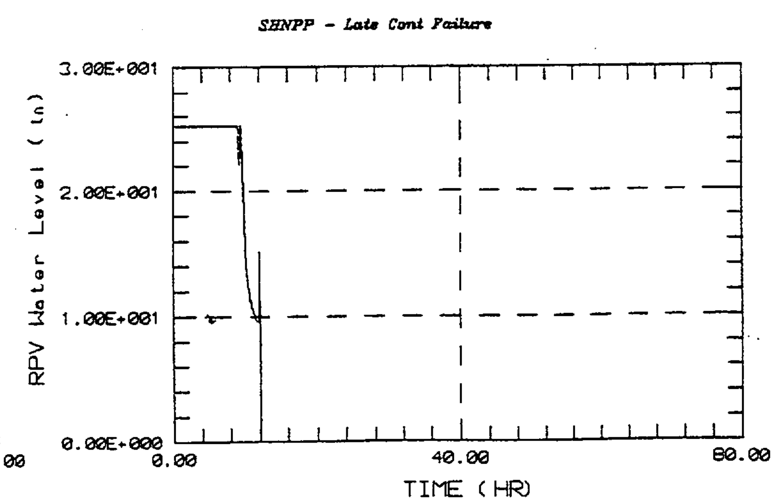
2.00E+005
1.00E+005
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
RPV Water Level (in)
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1.00E+001
0.00E+000
0.00
40.00
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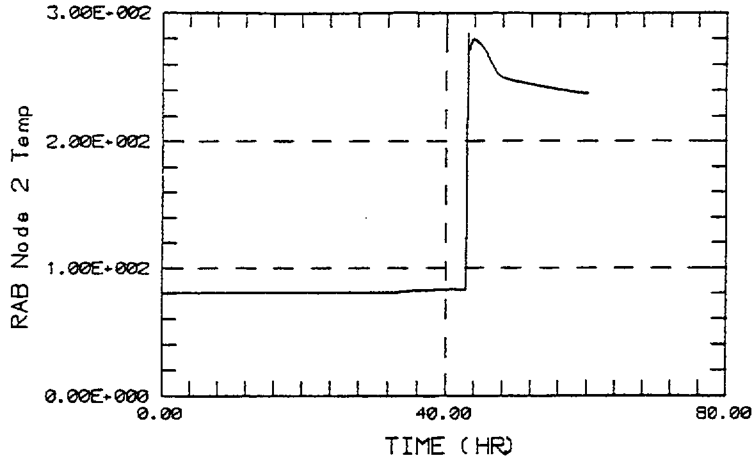
TIME (HR)

SHNPP – Late Cont Failure



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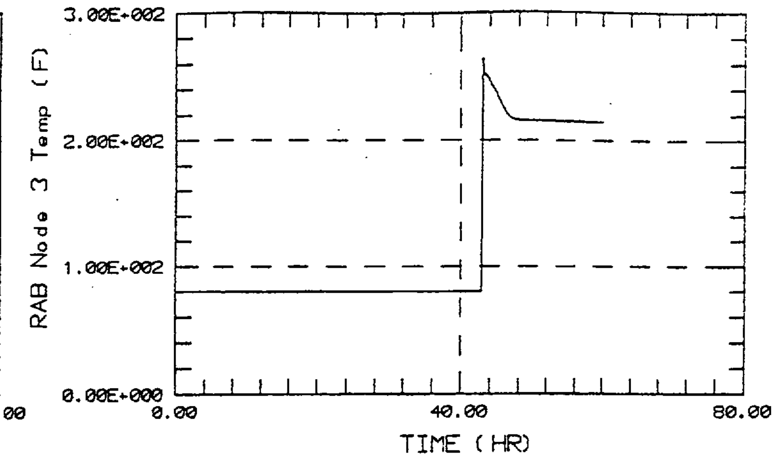

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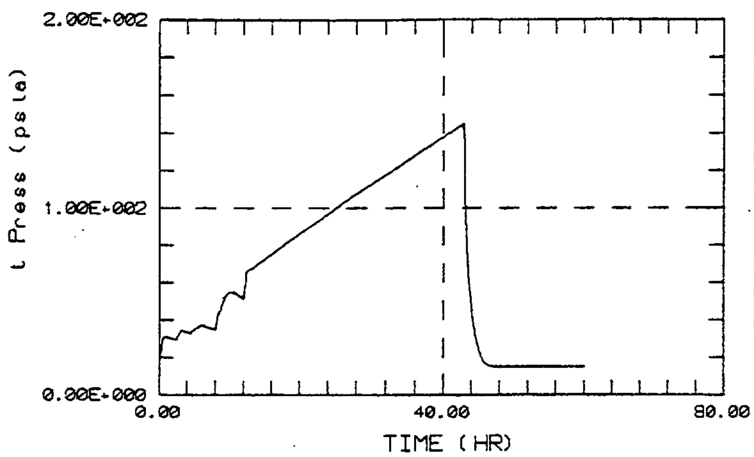

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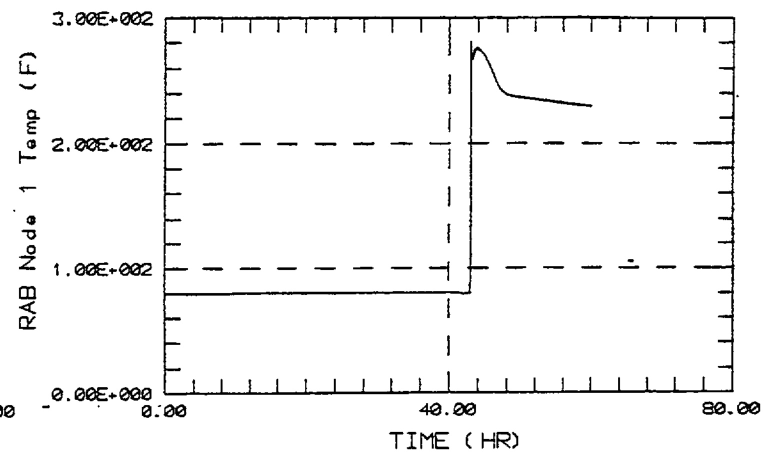

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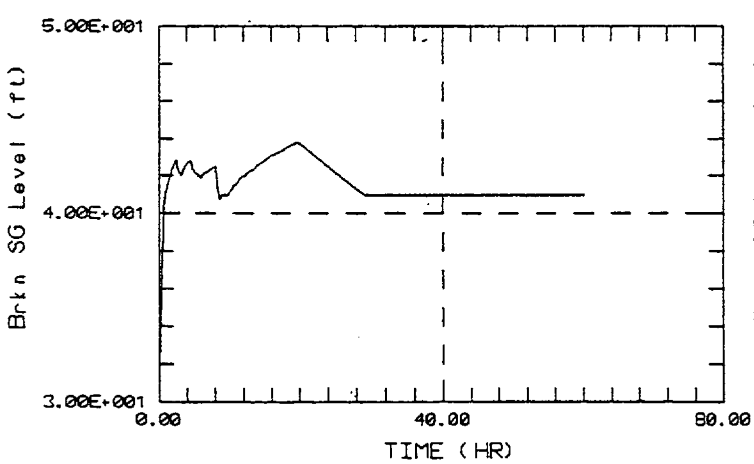
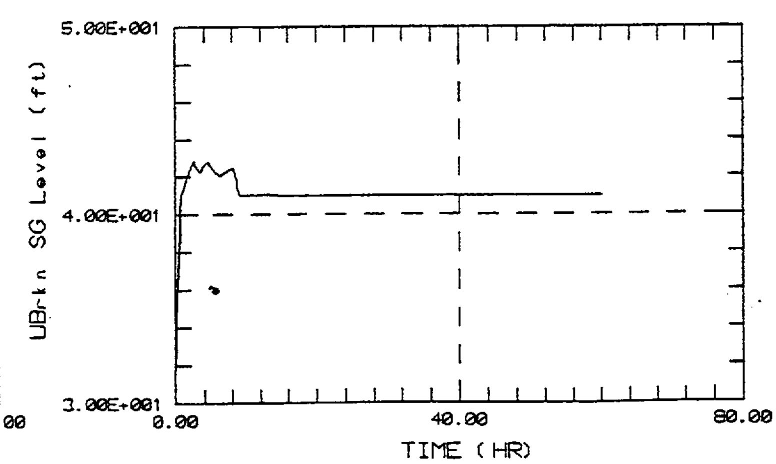
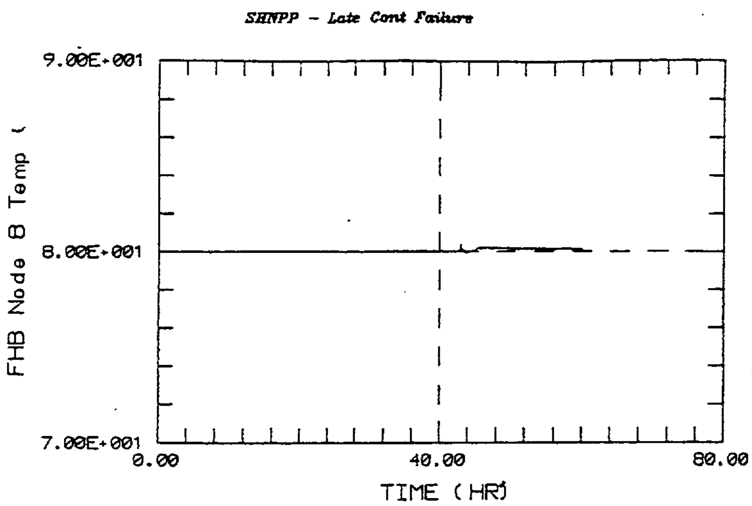
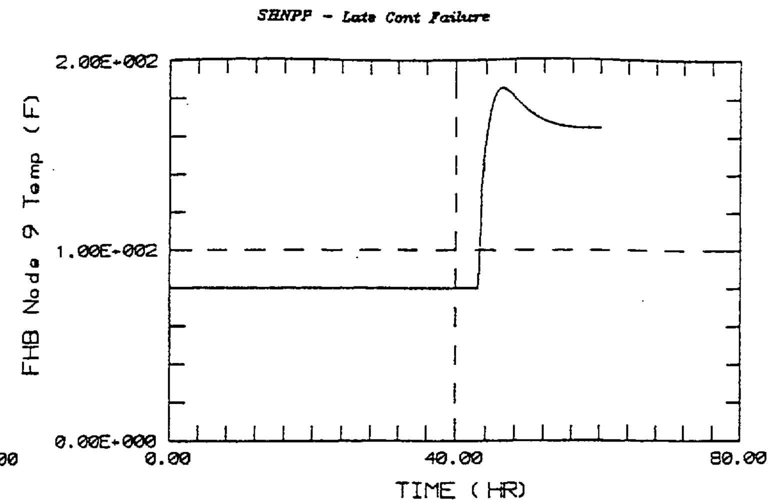
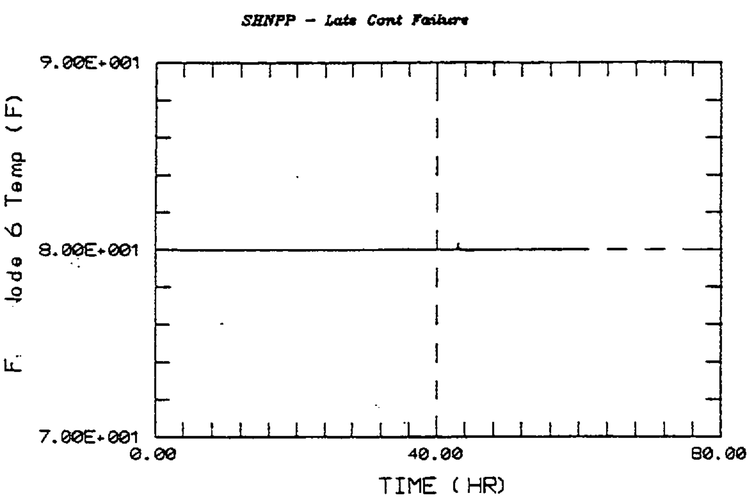


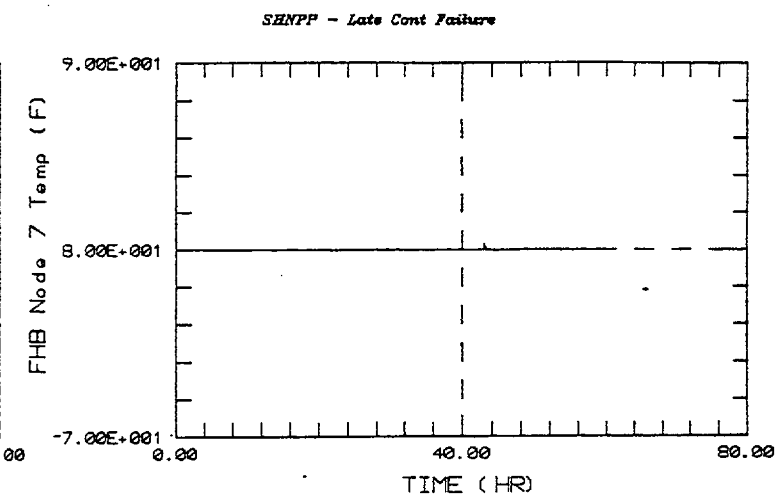
SHNPP – Late Cont Failure

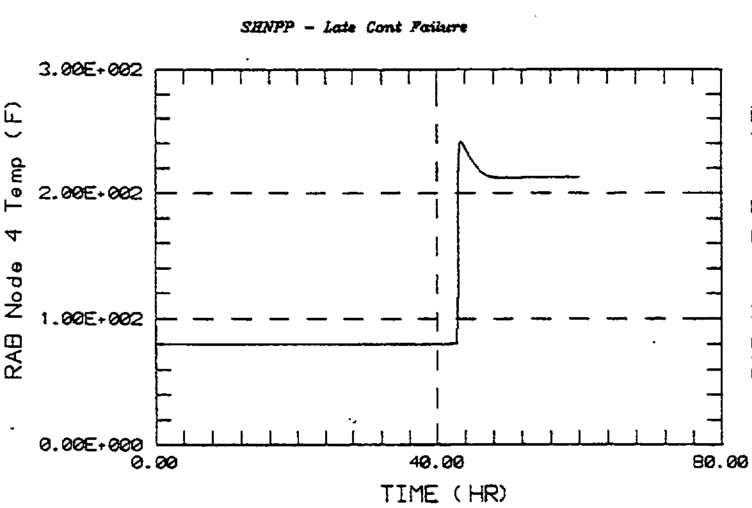


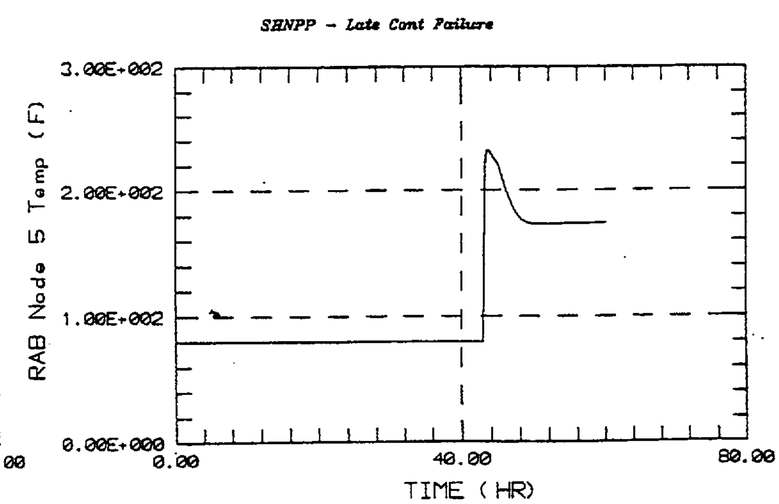

SHNPP - Late Cont Failure
9.00E+001
8.00E+001
7.00E+001
FHB Node 8 Temp
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
2.00E+002
1.00E+002
0.00E+000
FHB Node 9 Temp (F)
0.00
40.00
80.00
TIME (HR)

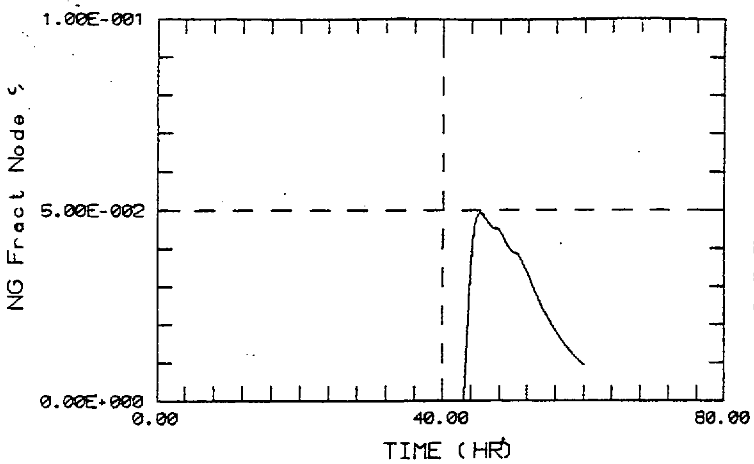

SHNPP - Late Cont Failure
9.00E+001
8.00E+001
7.00E+001
Node 6 Temp (F)
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
9.00E+001
8.00E+001
-7.00E+001
FHB Node 7 Temp (F)
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
3.00E+002
2.00E+002
1.00E+002
0.00E+000
RAB Node 4 Temp (F)
0.00
40.00
80.00
TIME (HR)


SHNPP - Late Cont Failure
3.00E+002
2.00E+002
1.00E+002
0.00E+000
RAB Node 5 Temp (F)
0.00
40.00
80.00
TIME (HR)

SHNPP – Late Cont Failure



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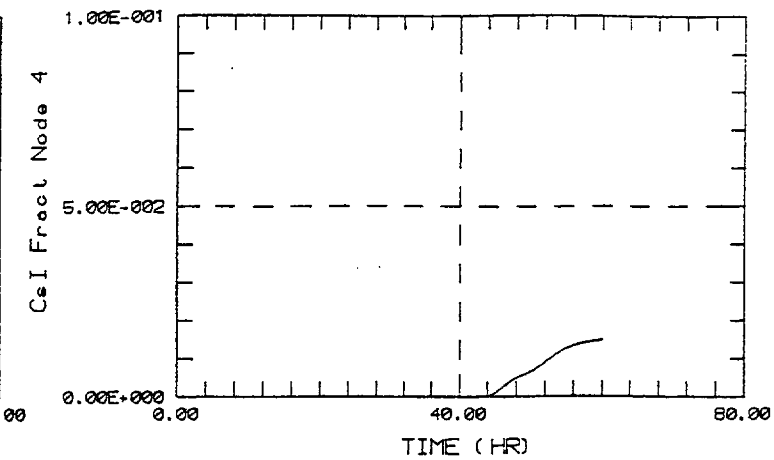

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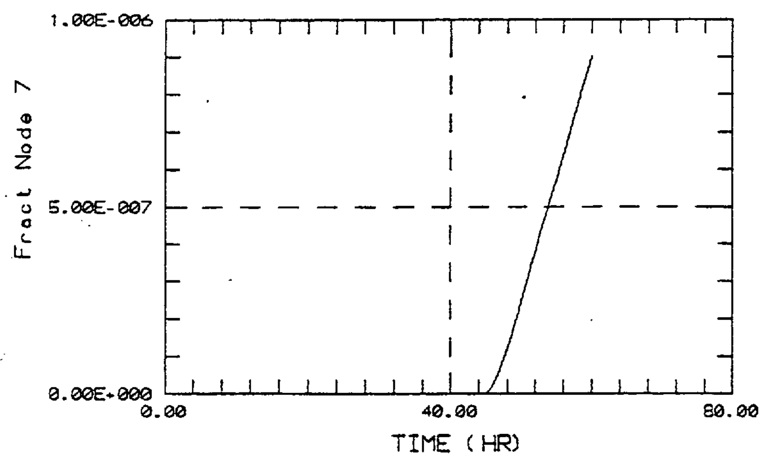

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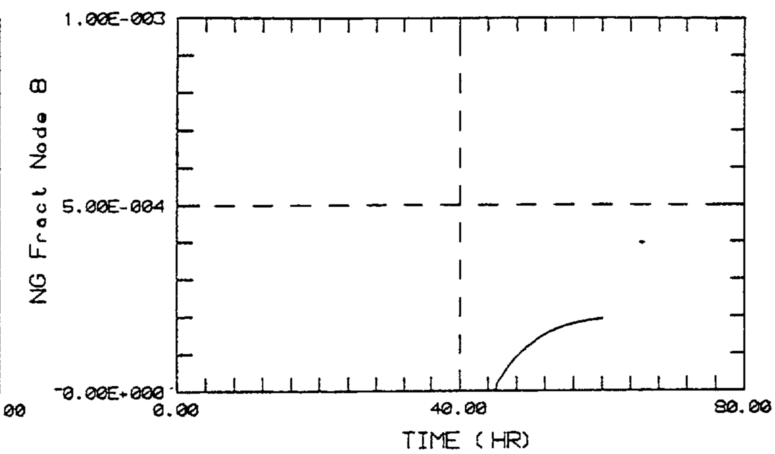
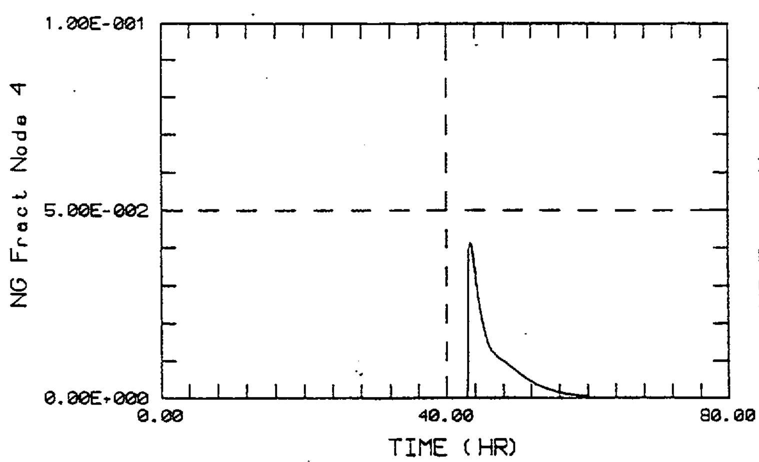
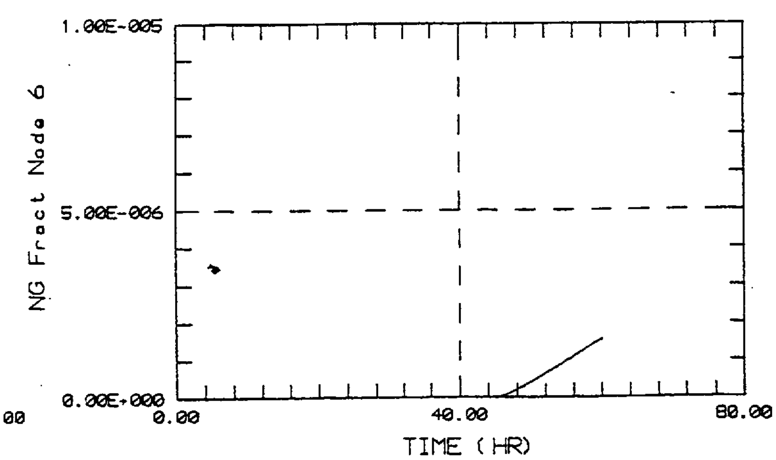
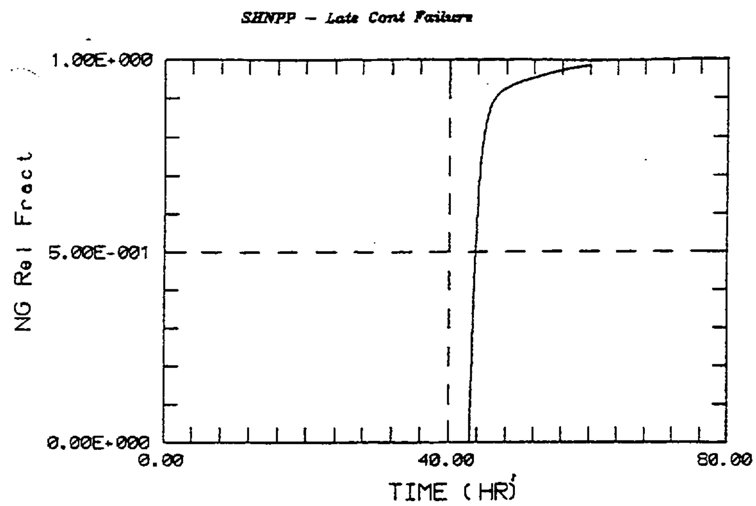


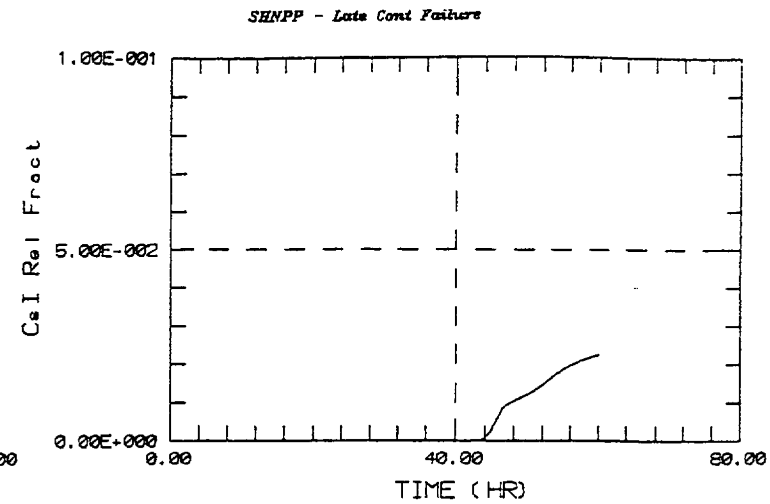
SHNPP – Late Cont Failure

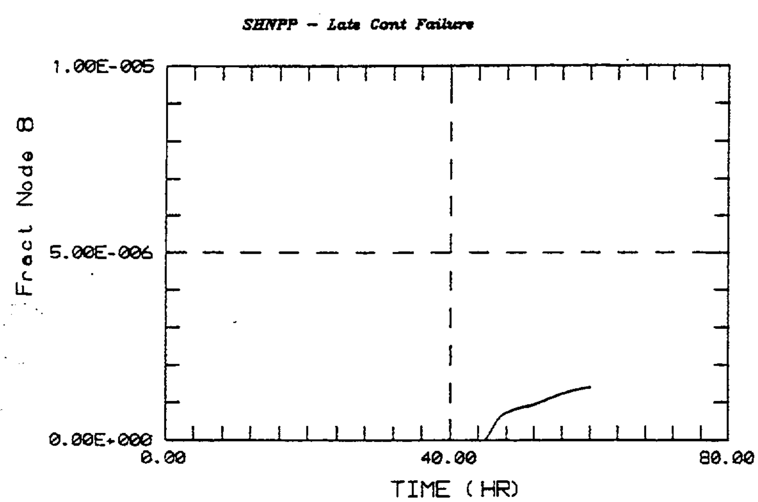


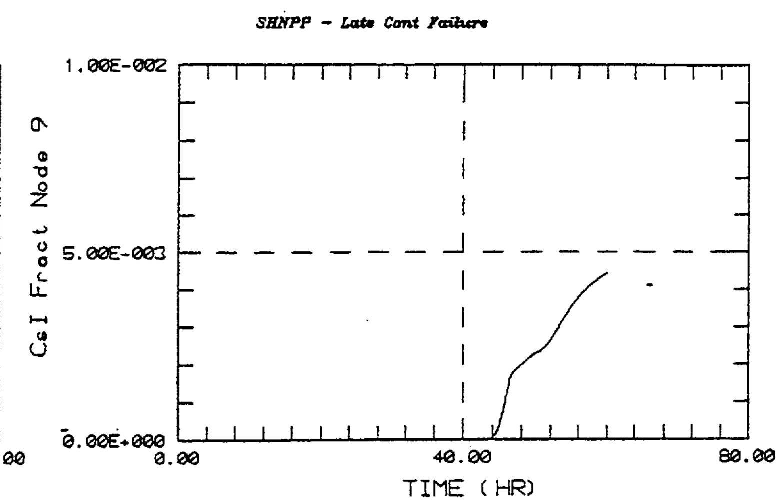
SHNPP – Late Cont Failure

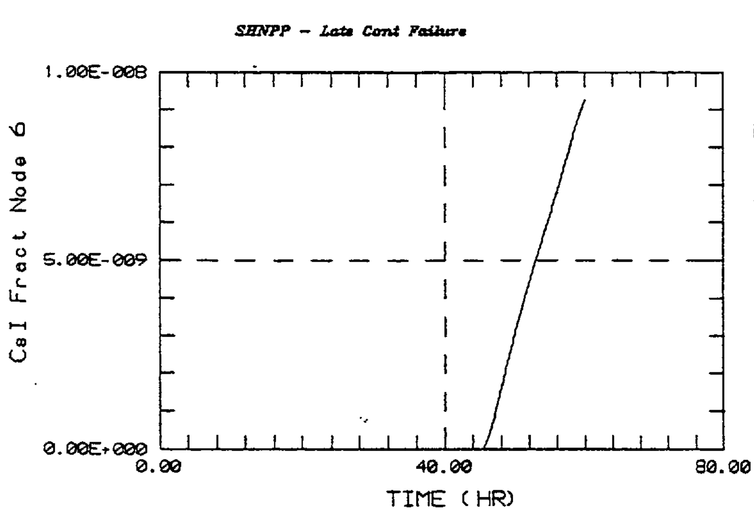


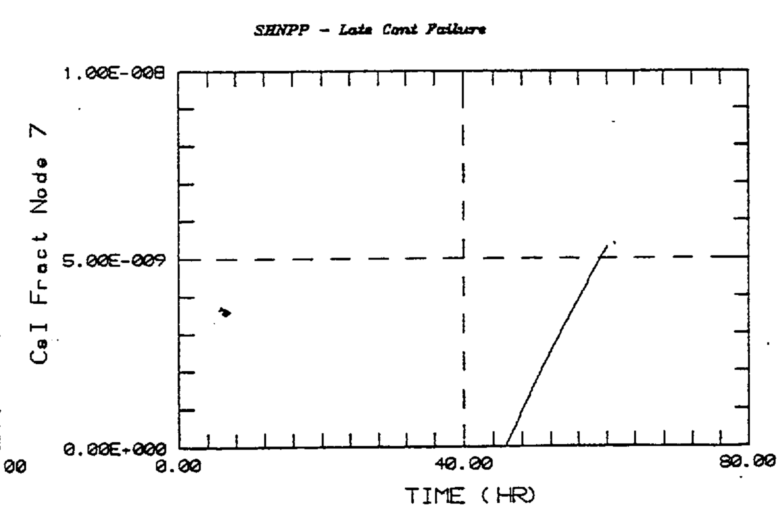

SHNPP – Late Cont Failure
NG Rel Fract
1.00E+000
5.00E-001
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP – Late Cont Failure
CsI Rel Fract
1.00E-001
5.00E-002
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP – Late Cont Failure
Fract Node 8
1.00E-005
5.00E-006
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP – Late Cont Failure
CsI Fract Node 9
1.00E-002
5.00E-003
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP – Late Cont Failure
CsI Fract Node 6
1.00E-008
5.00E-009
0.00E+000
0.00
40.00
80.00
TIME (HR)


SHNPP – Late Cont Failure
CsI Fract Node 7
1.00E-008
5.00E-009
0.00E+000
0.00
40.00
80.00
TIME (HR)

APPENDIX F

WALKDOWN OF THE SHEARON HARRIS REACTOR AUXILIARY AND FUEL HANDLING BUILDINGS

In support of the Shearon Harris Spent Fuel Pool Evaluation, two plant walkdowns were conducted on August 25 and September 21, 2000. Participating in the walkdowns were:

August 25, 2000	September 21, 2000
Eric McCartney (CP&L)	Steve Edwards (CP&L)
Steve Edwards (CP&L)	Joe Davis (CP&L)
Bruce Morgen (CP&L)	Edison Morales (CP&L)
Edison Morales (CP&L)	Ed Burns (ERIN)
Jeff Gabor (ERIN)	Tom Daniels (ERIN)
Tom Daniels (ERIN)	Jeff Gabor (ERIN)

The main objectives of the walkdowns were to observe potential gas flow paths connecting the Reactor Auxiliary Building (RAB) with the Fuel Handling Building (FHB) and to view the locations where Fuel Pool make-up could be aligned. The gas flow paths are important in the assessment of the radiological environment following severe core damage accident scenarios if containment is failed or bypassed. Details of the junctions connecting buildings and compartments are provided on the reference drawings (i.e., door swing direction) and will not be repeated in this report. However, confirmation of important junctions was obtained during the walkdown.

The following provides notes and observations related to the walkdowns.

Reference Material

General Arrangement Drawings for FHB and RAB – CAR-2165
Drawings G-015 through G-021
Drawings G-022 through G-026

Fuel Handling Building (FHB)

<u>El. 286' Operating Deck</u>

Refer to Figures F-1 through F-3 for general photos of the area. The operating deck volume is estimated at over 1 million cubic feet with solid concrete walls and ceiling. Normal controls for the spent fuel pool cooling operation are located on this elevation. At this elevation, the gas flow pathways were:

HVAC ducts near the ceiling elevation (see Figures F-4 & F-5) connecting to the RAB
Equipment hatches (2) connecting to El. 261' of the FHB (approx. 8' X 10')

Double doors (air lock) leading to south stairwell
Open stairwell to El. 261' of FHB
Open stairwell to fuel unloading area (261')
Equipment hatch (cover on – Figure 30) to fuel unloading area (261')
Elevator to fuel unloading area (261')

Additional photos of this area are included in Figures F-24 through F-29. Figures F-24 and F-25 show the 286' elevation of the FHB and the Fuel Pools located there. Figures F-26 through F-28 show the gates between the Fuel Pool B and the transfer canal. Figure F-28 shows the height of the installed gate relative to the normal water level and to the 286' elevation floor. .

Figure F-37 is a simplified sketch of the Fuel Pools. Note that the sketch is not to scale and the distance from Fuel Pool B to C is much larger than indicated.

El. 261' Fuel Unloading Area (North end of FHB)

This area communicates to the FHB El. 286' via an open stairwell and the equipment hatch (cover on). A large door exists for rail entrance from outside. There is also an air lock door to the outside for personnel access. This door was indicated to be a "tornado door".

El. 261'

This elevation contains two separate regions, one centrally located and the other at the north end. There is no communication pathway directly between these two regions on the 261' elevation. The central region includes the FHB Emergency Exhaust System. Access to this region is from the RAB through a door. This connection is into the area just below the exhaust air plenum and adjacent to the HVAC room in the northwest corner of the RAB. From this elevation communication to the 286' elevation is through 2 equipment hatches approximately 10' by 10' in dimension. These hatches were viewed from the 286' elevation and appear to be screwed into the floor of the FHB operating deck. There are also a corresponding set of hatches in the floor at 261' leading down to the 236' elevation of the FHB. These hatches appear to be not secured and sitting flush with the floor over the opening. Small leak areas were observed through the hatches at the hand hold locations.

There is also an area on the north end of the building that contains various decontamination facilities. There is an open stairwell leading up to the 286' elevation from this region.

El. 236'

Like the 261' elevation, this floor level contains 2 separate regions. The centrally located region contains the fuel pool heat exchangers and pumps. This elevation also contains local manually operated valves to use in the RWST-to-spent fuel pool line up along with a secondary set of controls for spent fuel pool cooling. On the west wall are the fuel pool cooling pumps along with a tornado door to the outside. Access to this region is via doors from the RAB (Figure 8). There are also 2 hatch covers installed leading down into the 216' elevation. The hatch on the south end of this region appears to sit over the opening, however, the hatch on the north end had a locking bar in place. (Figure F-22)

On the north end of the FHB is a separate decontamination area. There is a stairwell (air lock) on the north end leading down to El. 216' and up to El. 261' (Figure F-35). There is also a tornado door allowing direct entry from outside the FHB. (from the Safety Meeting Room area (Figure F-34). Access to the 216' elevation which contains valve 2SF201 and C&D purification pumps is by : (a) using the ladder mounted on the wall next to the stairwell (Figure 35) through the opening in the floor (Figure F-36); or (b) the closed stairwell from 236 El. To 216' El.

El. 216' (Basement of FHB)

This is the basement of the FHB. There are also 2 separate regions at this elevation. On the South end of the FHB are the filter backwash transfer pumps, tanks, and various leak detection stations. This region also includes the Pool A&B fuel pool purification pumps (Figure 16). There is a equipment hatch (Figure F-15) leading up to the 236' elevation as described previously. The equipment hatch has hand hole grips that allow leakage through the hatch.

There is also a door to the RAB in the area of the aux steam condensate tank.

There is a similar region on the North end of the FHB for the Pool C&D purification pumps (Figures F-31 through F-33). Access to this region is from a stairwell from the 236' elevation. The other equipment hatch is located in this region leading up to the 236' elevation. The equipment hatch has hand hole grips that allow leakage through the hatch.

One of the methods of makeup to the Spent Fuel Pools is the use of the Demineralized Water System. The critical manual valves are located at El. 216'. The manual valve to the Unit 1 purification system is located in the South compartment (1SF201) and the manual valve to the Unit 2 purification system is located in the North compartment (2SF201). The 1SF201 valve is shown on a piping layout in Figure F-38 as Number 45.The manual valves are padlocked closed (see Figure F-31). Emergency lighting is available in both areas to support operator actions. The Auxiliary Operator (AO) carry keys for the padlocks.

Reactor Auxiliary Building (RAB)

El. 337' Roof

There are 2 hatch covers for the RHR Heat Exchangers. Refer to Figure F-9.

Exhaust stack for HVAC from FHB and RAB – Figure F-10.

El. 324'

This area contains HVAC supply and exhaust systems with ducts leading into the operating deck of the FHB (Figure F-23).

El. 261'

Southwest corner – large double door to the Waste Processing Building (WPB) – Figures F-6 & F-7

"Tornado Door" access (2) to steam tunnel

On the north end of the RAB is access to the FHB (Figure F-8). There are two doors located here that allow communication with the FHB.

El. 236'

There is good communication via an open pipe tunnel all the way down to El. 190'.

ESW-to-fuel pool makeup is established at this elevation, i.e., local valve manipulations are required for ESW alignment for fuel pool makeup. This is located just outside the hot machine shop doorway (Figure F-21). The ESW hose connections are located in the compartment overhead approximately 20 ft. above the floor. The "gang box" containing the required ESW hose connections and wrench is located in a separate compartment on this elevation.

El. 216'

This would likely be the release elevation for the containment wall-to-floor failure location as identified in the Harris IPE. It is also assumed to be the location for basemat failure (this may be conservative).

Door opening from RAB to WPB at West edge of containment (Figure F-11). This allows the WPB to communicate with the NW corner of the RAB. In this quadrant of the RAB, there is a set of double doors leading into the FHB (Figure F-12). There is also a doorway eading into the NE quadrant of the RAB (Figure F-13 and F-14).

NE Corner Hatch cover 4'X4' in place over ladder from El. 190' (Figure F-17), i.e., to the RHR compartment

NE Corner Ladder up to El. 261'

6" Gap around pipe penetration to El. 190' (Figure F-18)

Figure 19 shows the concrete hatch plugs leading down to the 190' elevation.

El. 190'

RHR and Containment spray pumps located on this elevation.

General Observations

For hypothetical breaks into the RAB, the flow paths identified in the walkdown will need to be assessed for their opening pressures. Doorways and equipment hatch covers will need to be analyzed as possible junctions for flow into the WPB, FHB, and environment. A key part of the assessment will be to establish what portion of the break flow is discharged into the FHB. The FHB is a place where local manual actions to perform recovery actions are possible. While the FHB represents a very large volume it also represents the potential for holdup and removal of any fission products released either from the primary system or from the spent fuel pool. This could adversely impact operator actions in the FHB, if a pathway from the RAB to the FHB is opened.

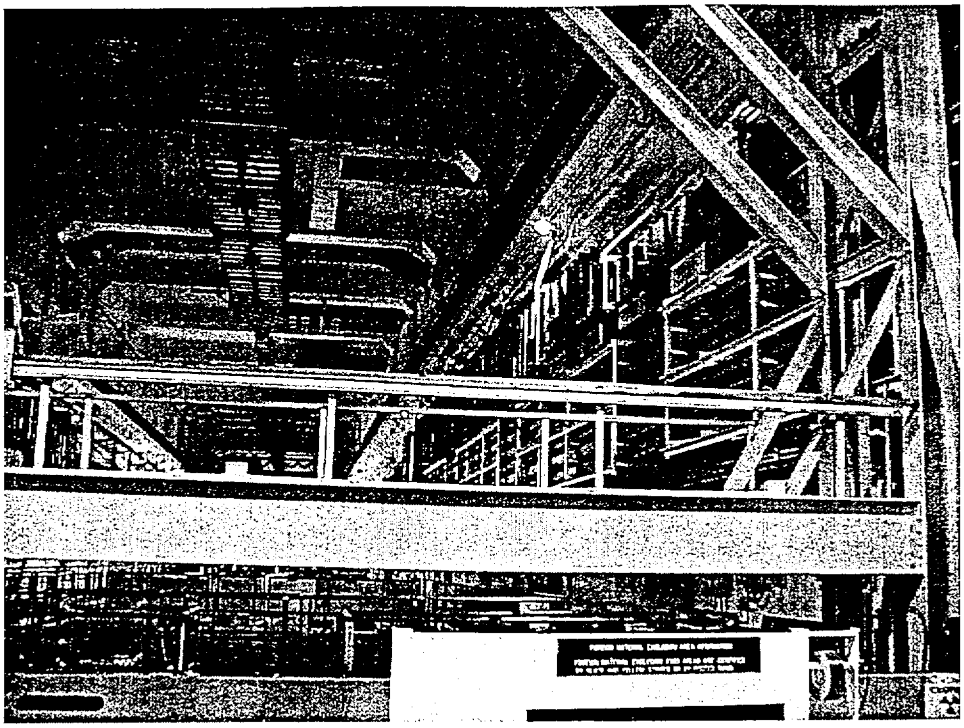


Figure F-1 – FHB El. 286'

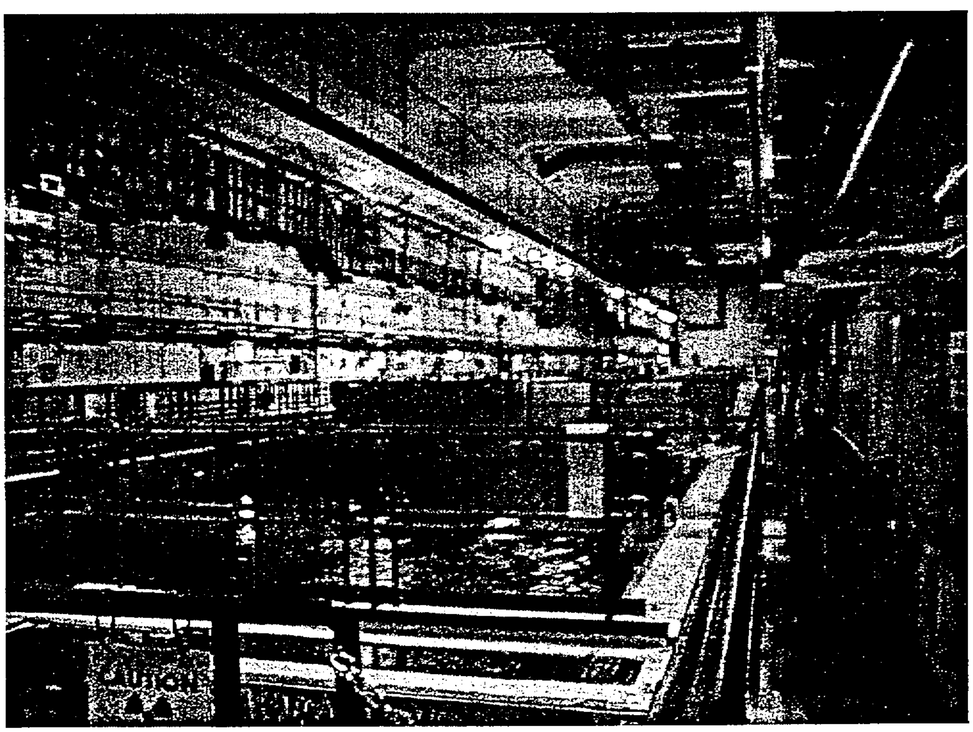


Figure F-2 – FHB El. 286'

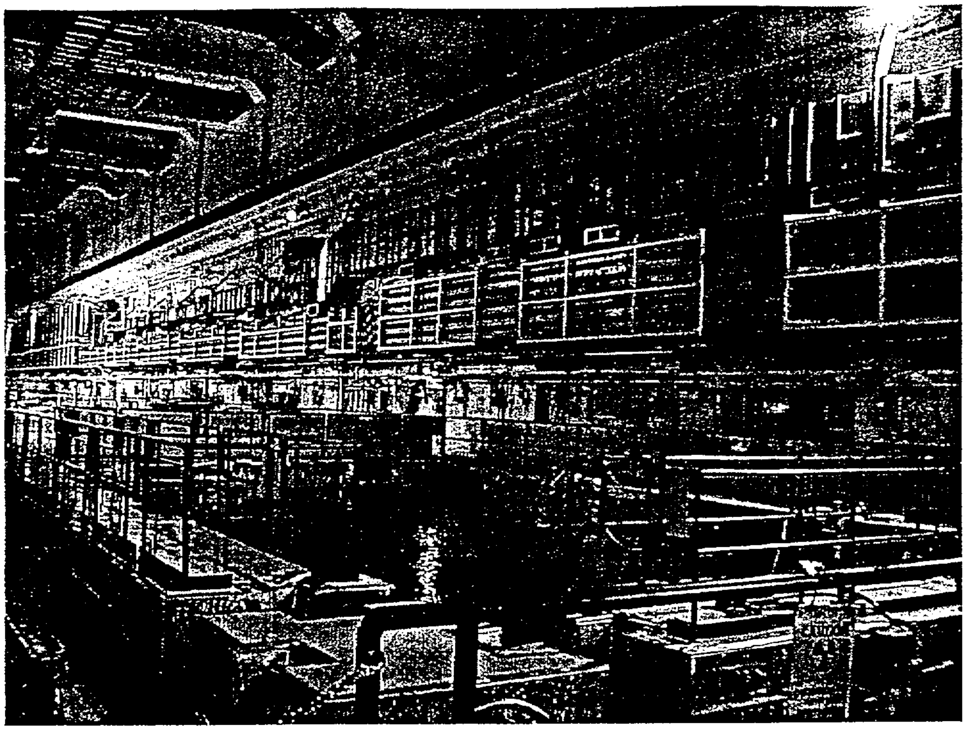


Figure F-3 – FHB El. 286'

Figure F-4 - FHB HVAC Penetrations

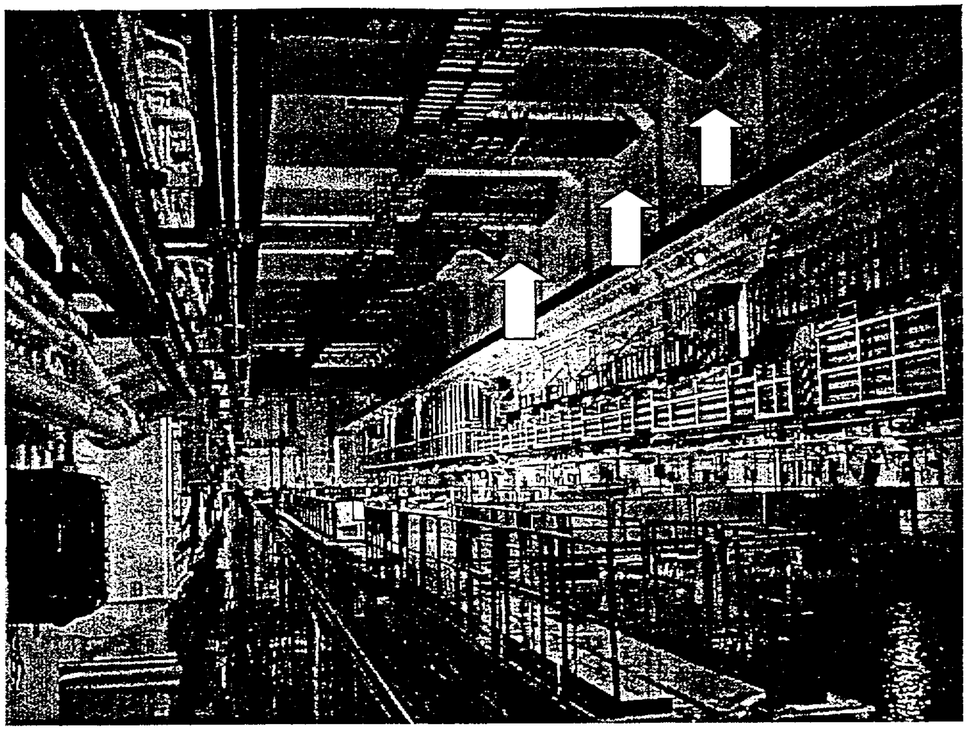


Figure F-5 – FHB HVAC Penetrations

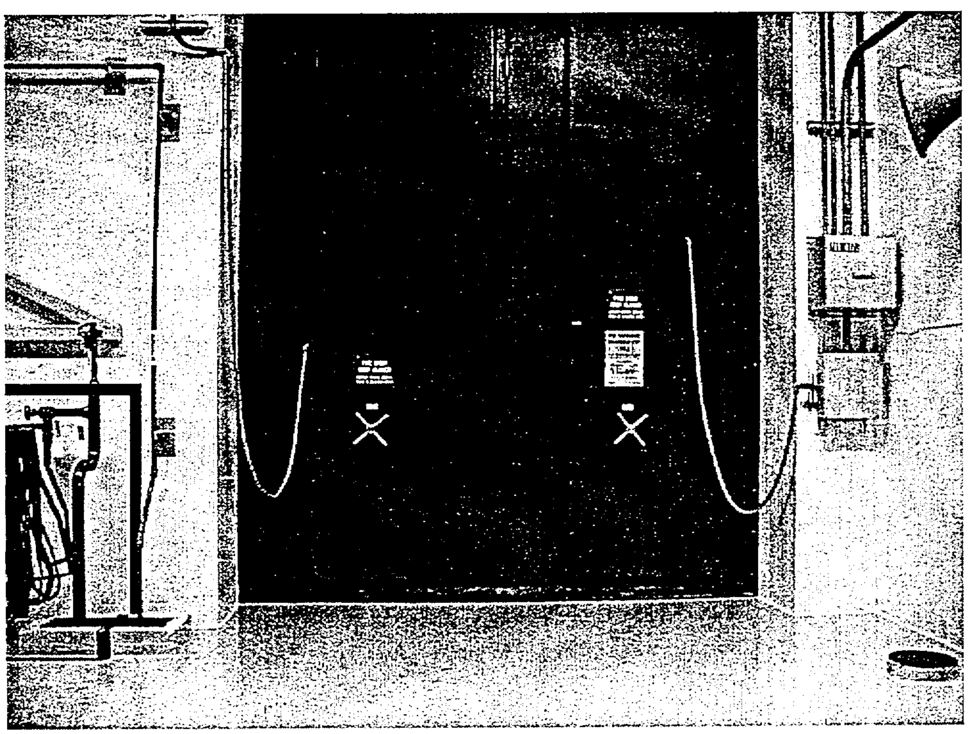


Figure F-6 – WPB-to-RAB door El. 261' (taken in RAB)

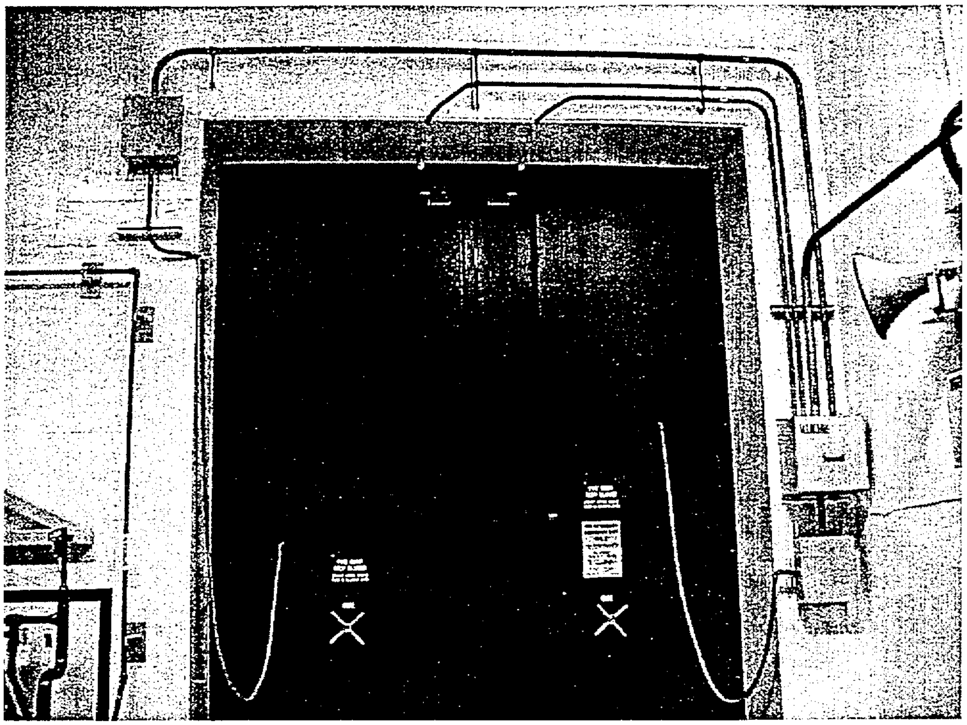


Figure F-7 – WPB-to-RAB door El. 261' (taken in RAB)

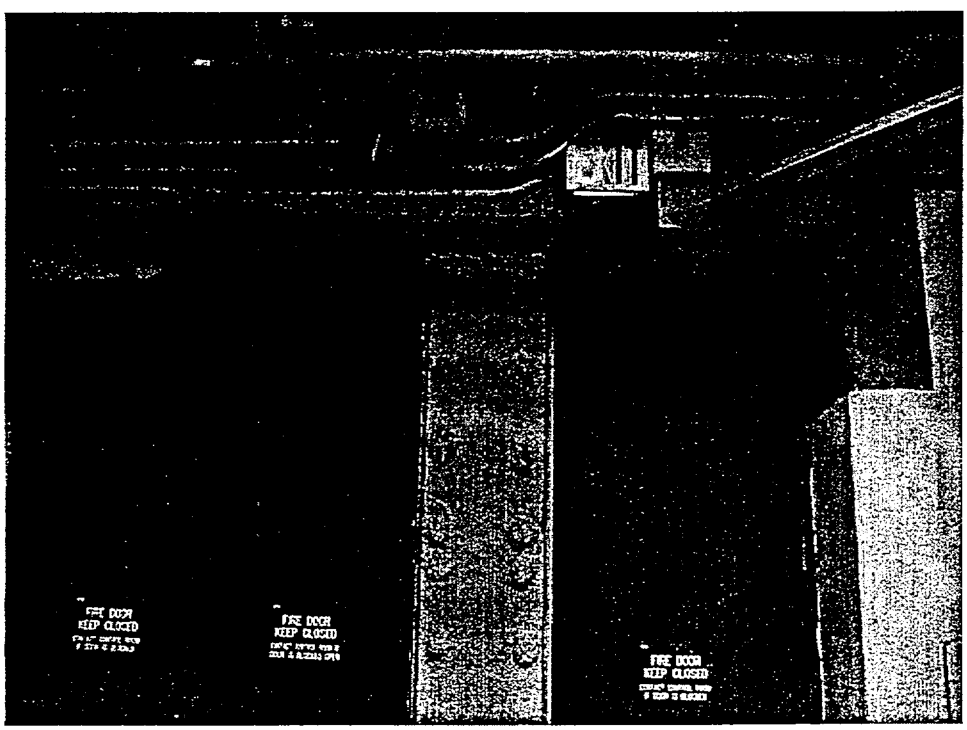


Figure F-8 – RAB-to-FHB doors El. 261' (taken in RAB)

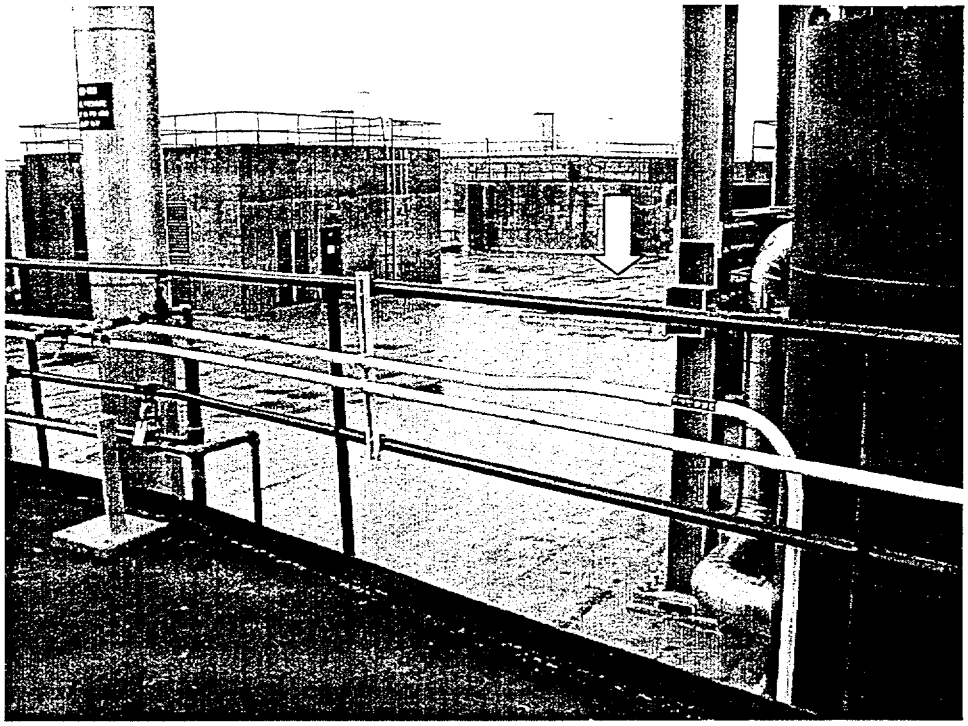


Figure F-9 – RHR Hatch Cover

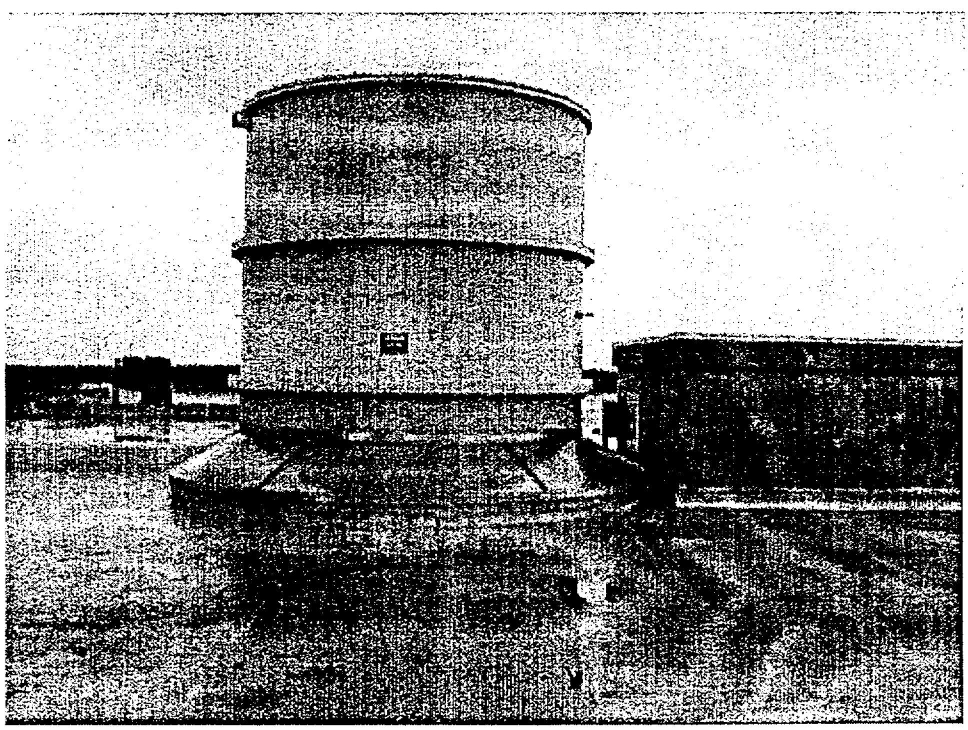


Figure F-10 – HVAC Exhaust Stack

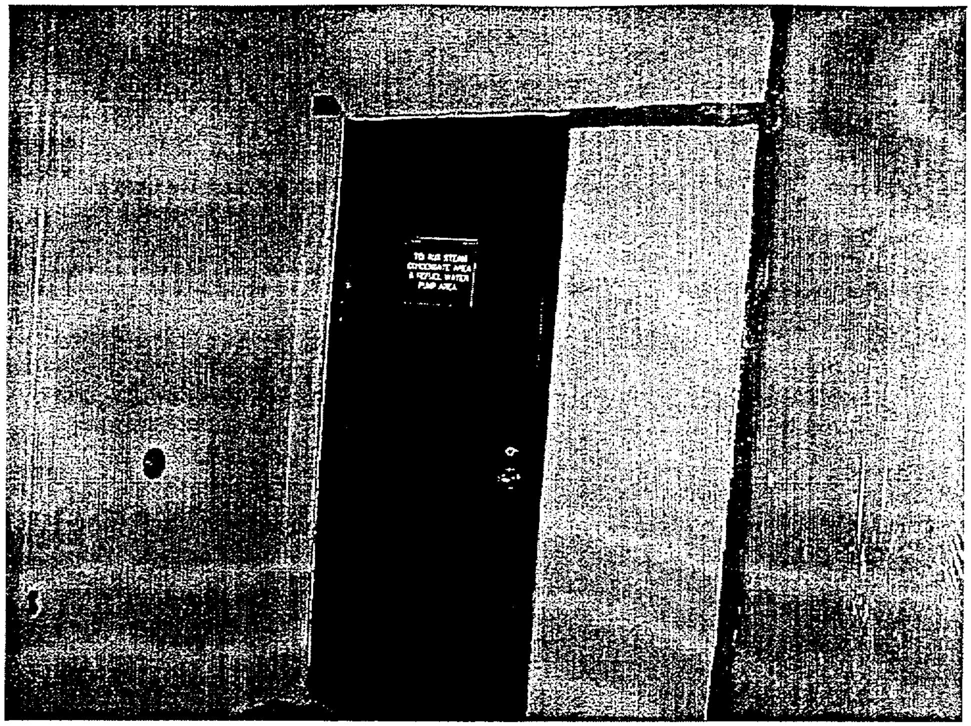


Figure F-11 – WPB El. 216' Door – Containment West Edge, From RAB to WPB

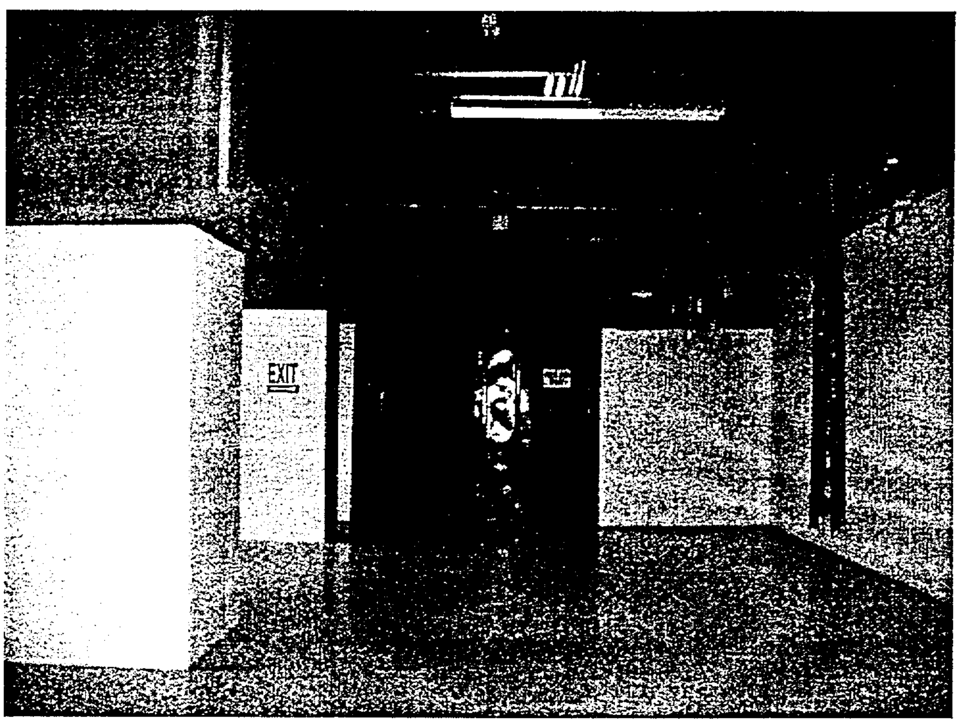


Figure F-12 – RAB El. 216' NW Corner Door, Opens from RAB into FHB

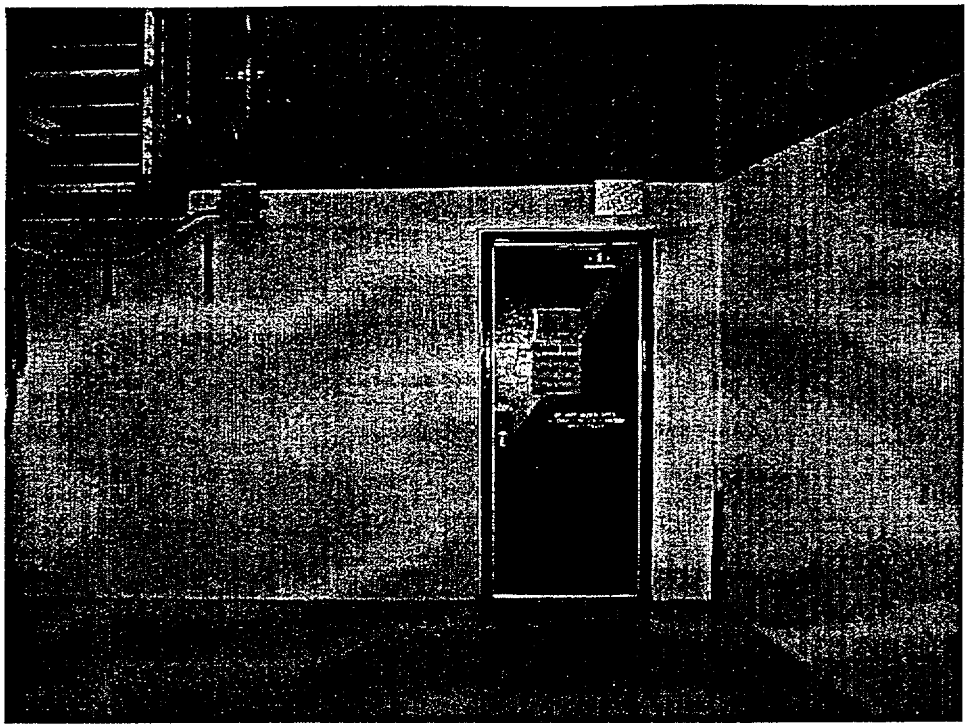


Figure F-13 – RAB El. 216' N Door Opens into NW Quadrant

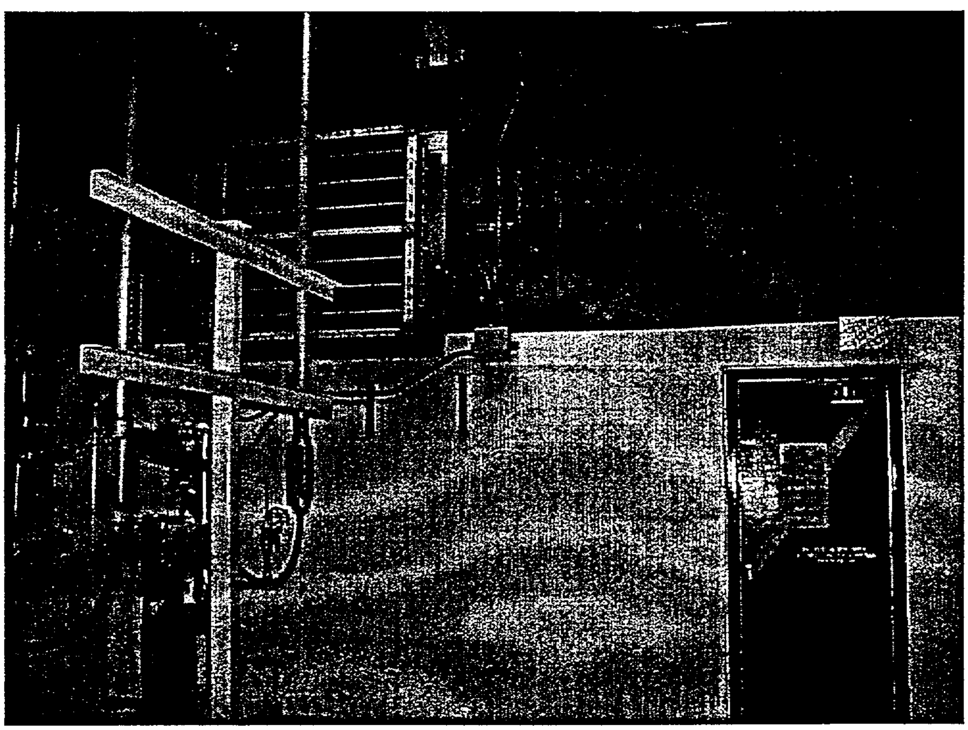


Figure F-14 – RAB El. 216' N Door Opens into NW Quadrant

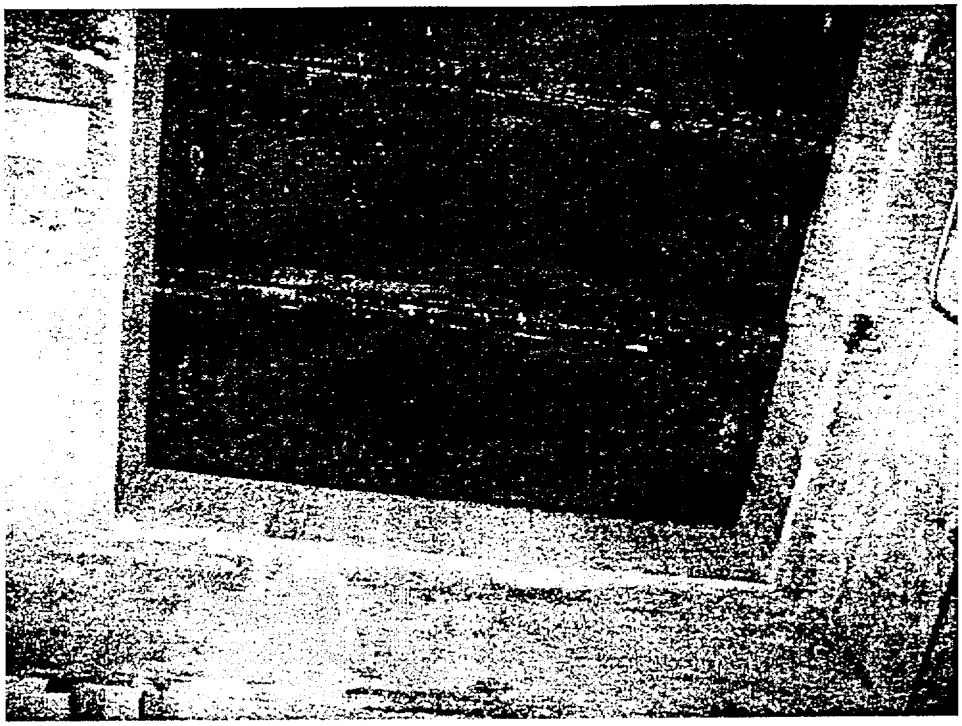


Figure F-15 – FHB El. 216' S, Hatch above doorway to El. 236'

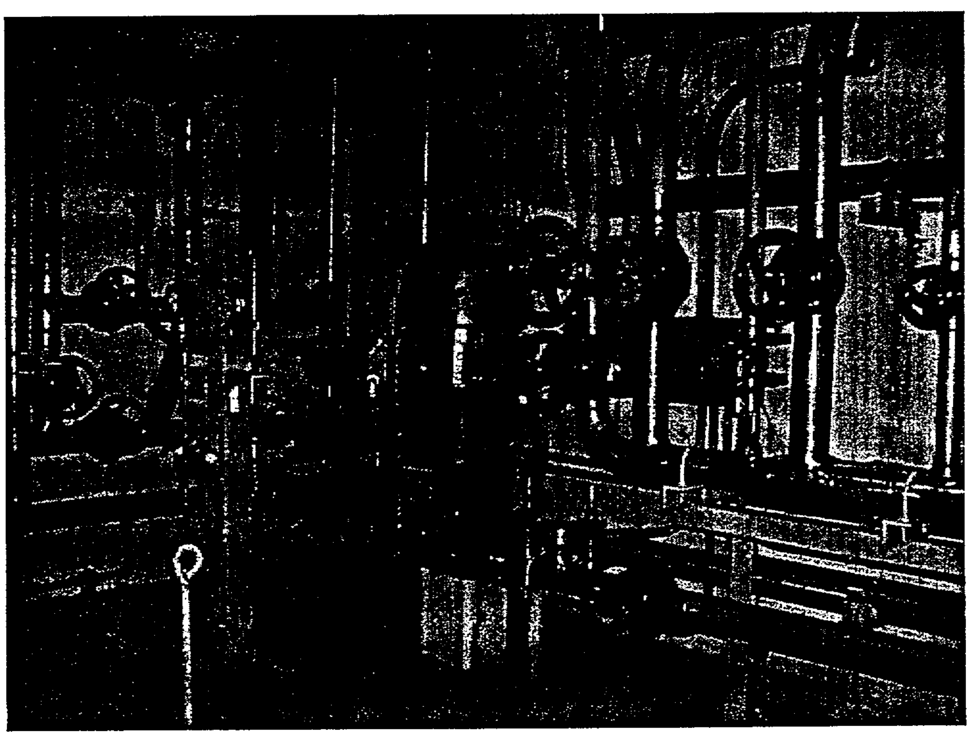


Figure F-16 – FHB El. 216' S, Fuel Pool Purification Pump

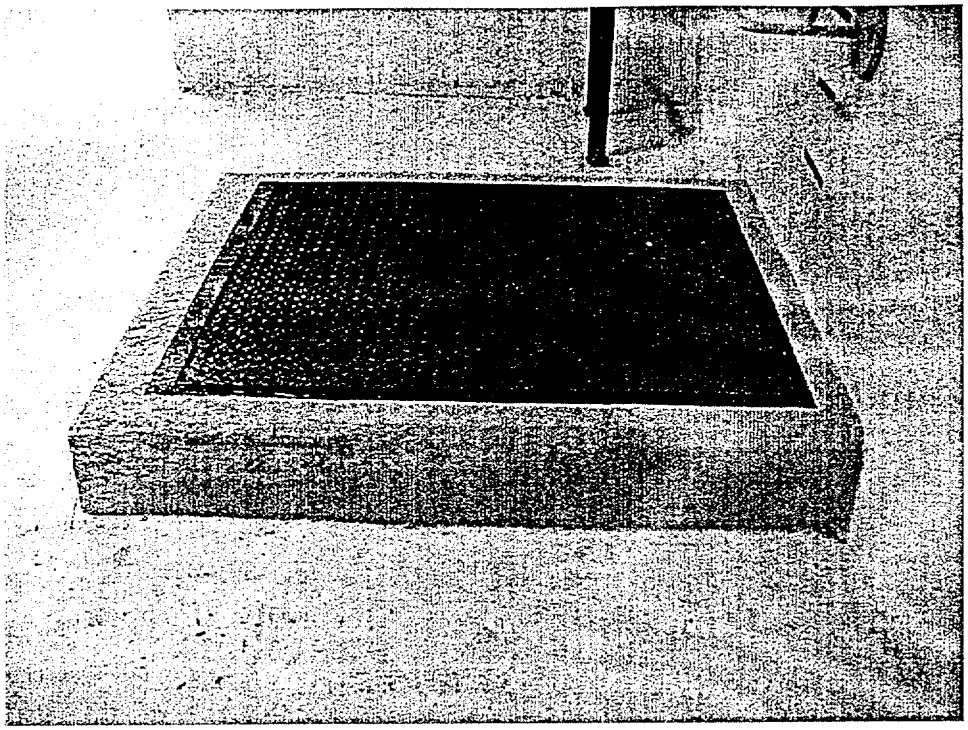


Figure F-17 – RAB El. 216' SE, Hatch down to El. 190'

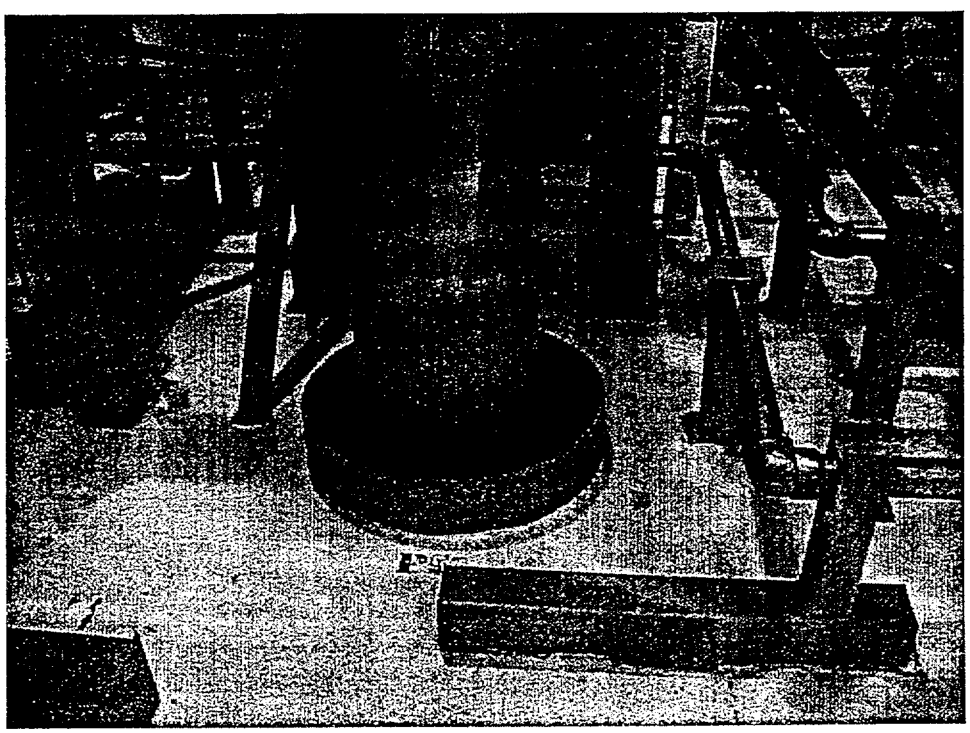


Figure F-18 – RAB El. 216' W, Gap around pipe to El. 190'

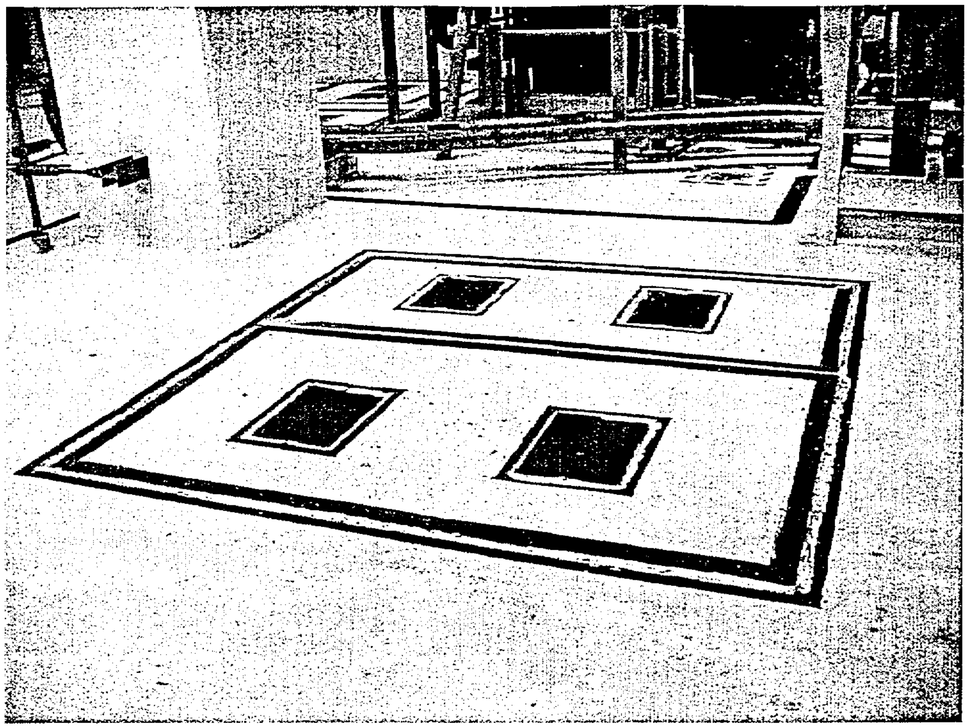


Figure F-19 – RAB El. 216' NE, Concrete Hatch Plug to El. 190'

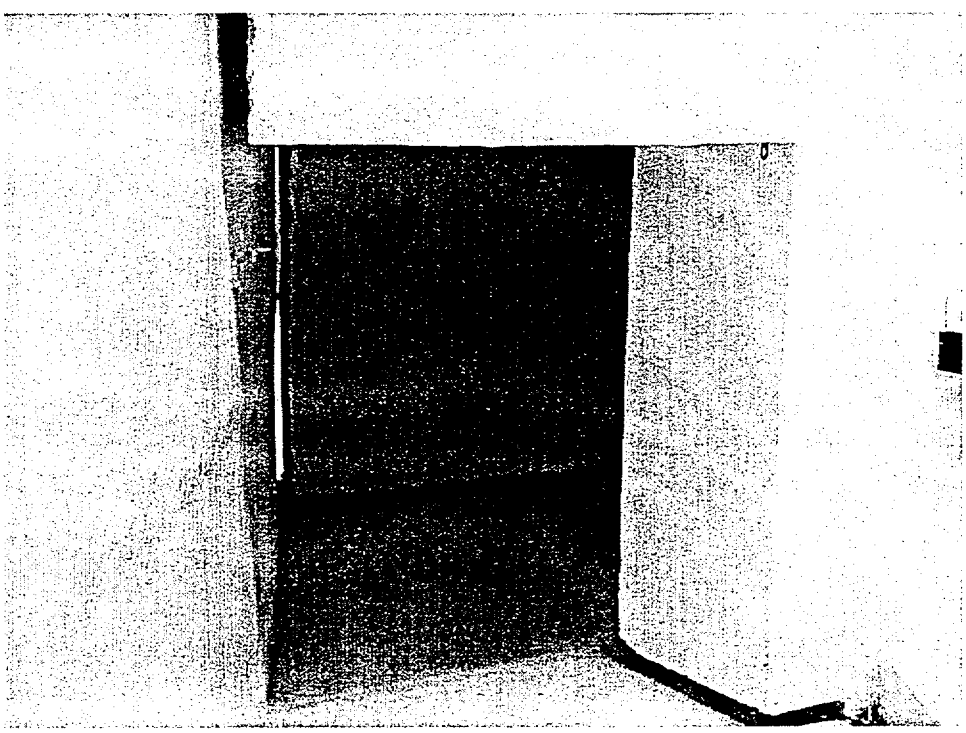


Figure F-20 – RAB El. 216' N, Inspection Access Around Floor Opening

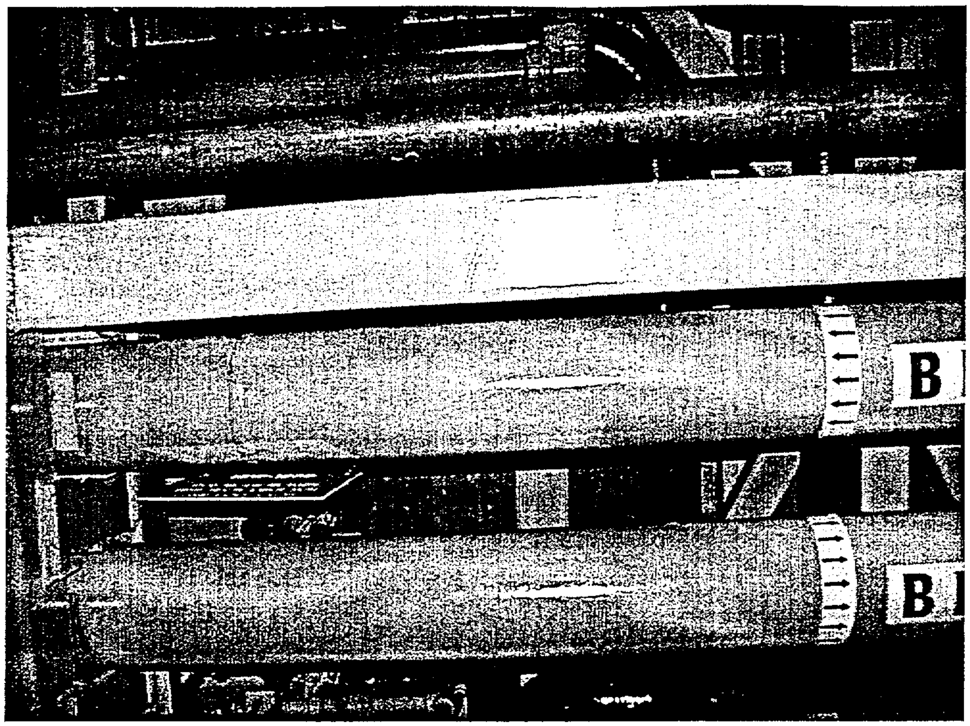


Figure F-21 – RAB El. 236' NE, 1-SW-1239, Near Hot Mach Shop

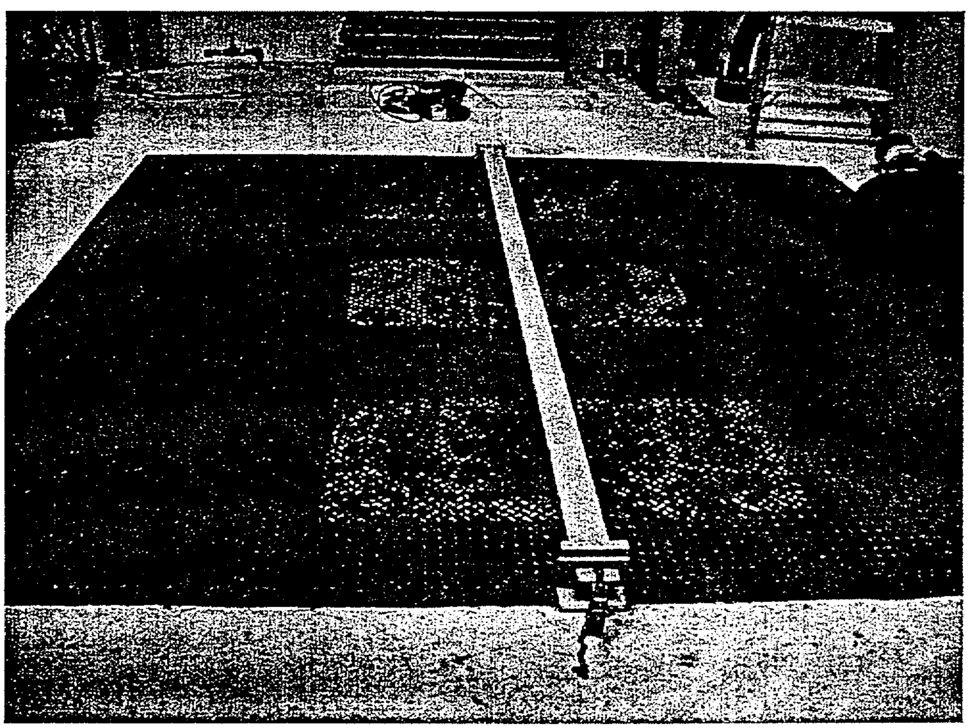


Figure F-22 – FHB El. 236' N, Lock on Hatch to El. 216'

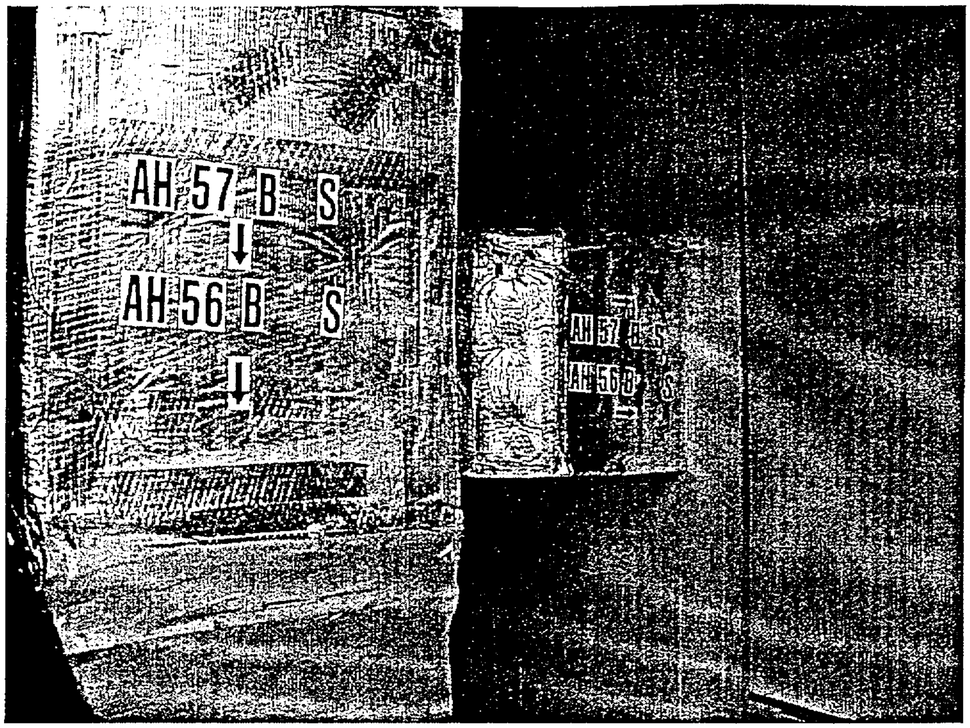


Figure F-23 – RAB El. 324' HVAC Supply to FHB El. 286'

Figure F-24 – FHB El. 286' Looking North

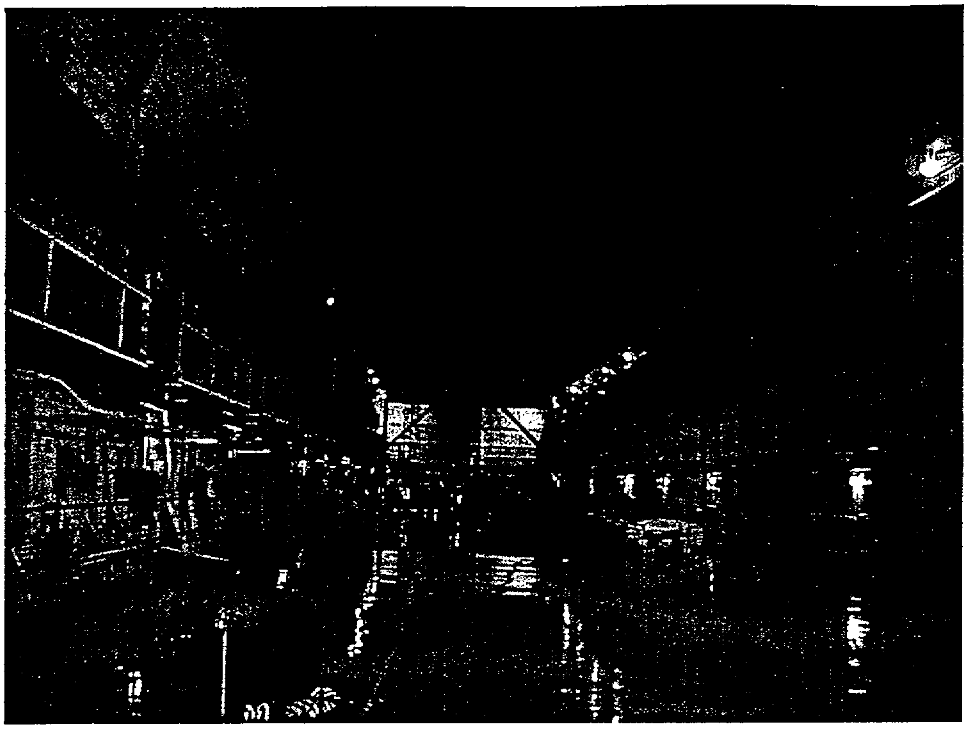


Figure F-25 – FHB El. 286' Looking South

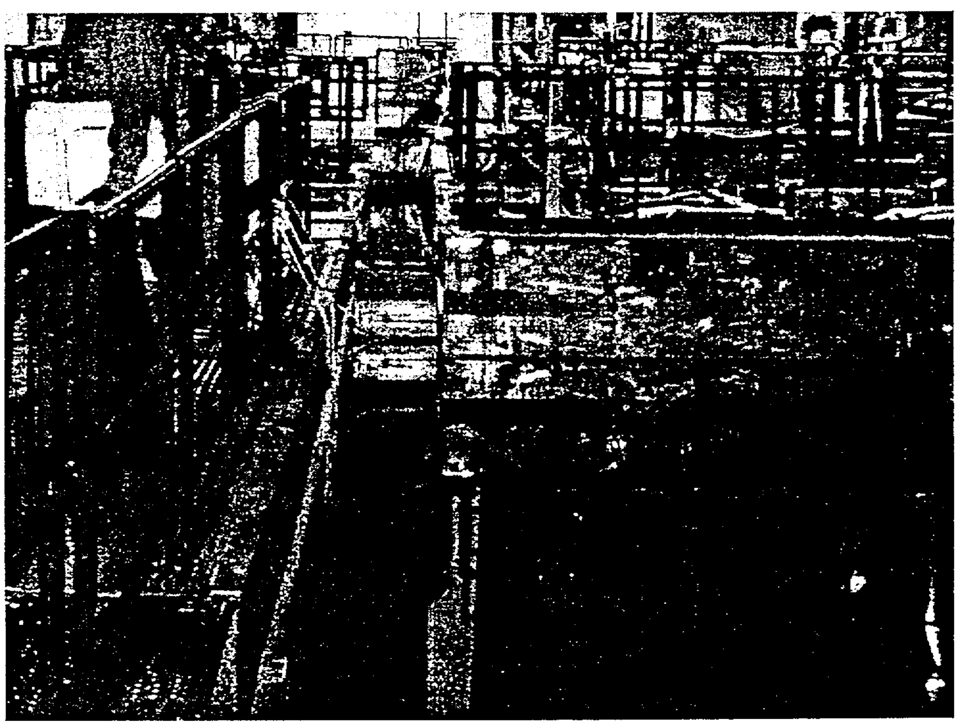


Figure F-26 – FHB El. 286' South End

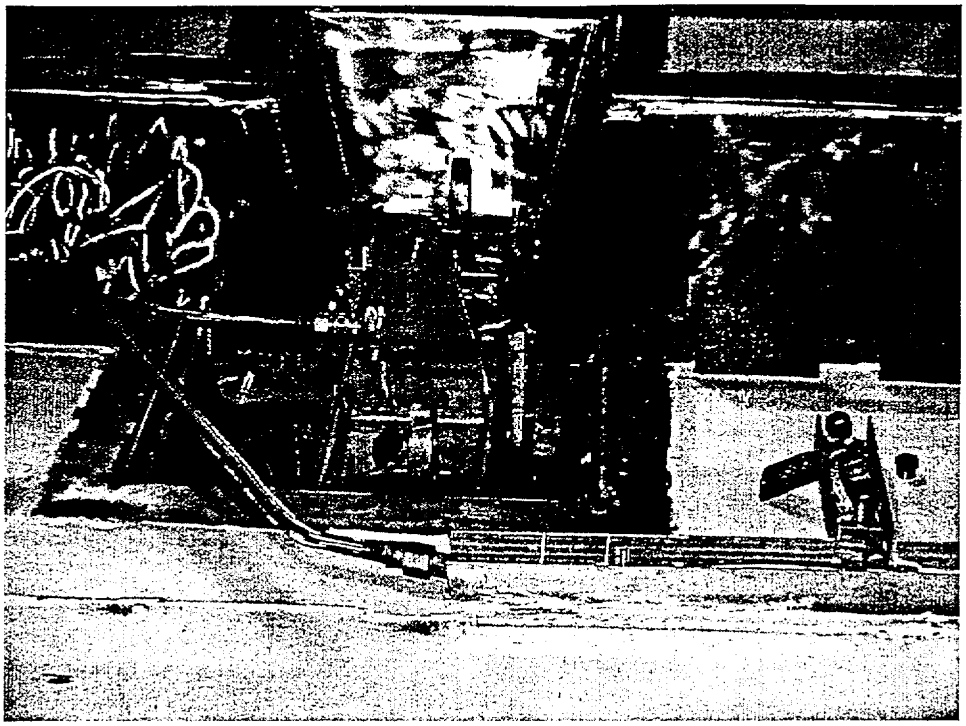


Figure F-27 – FHB El. 286' Bulkhead Gate 1SF-E0006

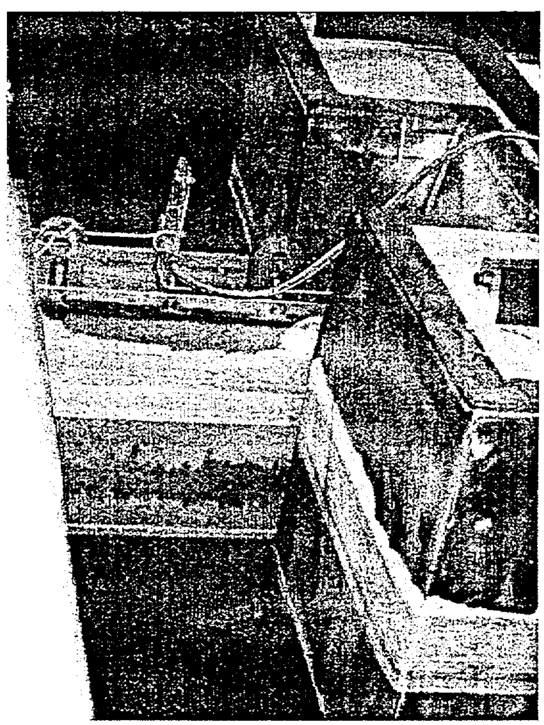


Figure F-28 – FHB El. 286' Bulkhead Gate 1SF-E0006

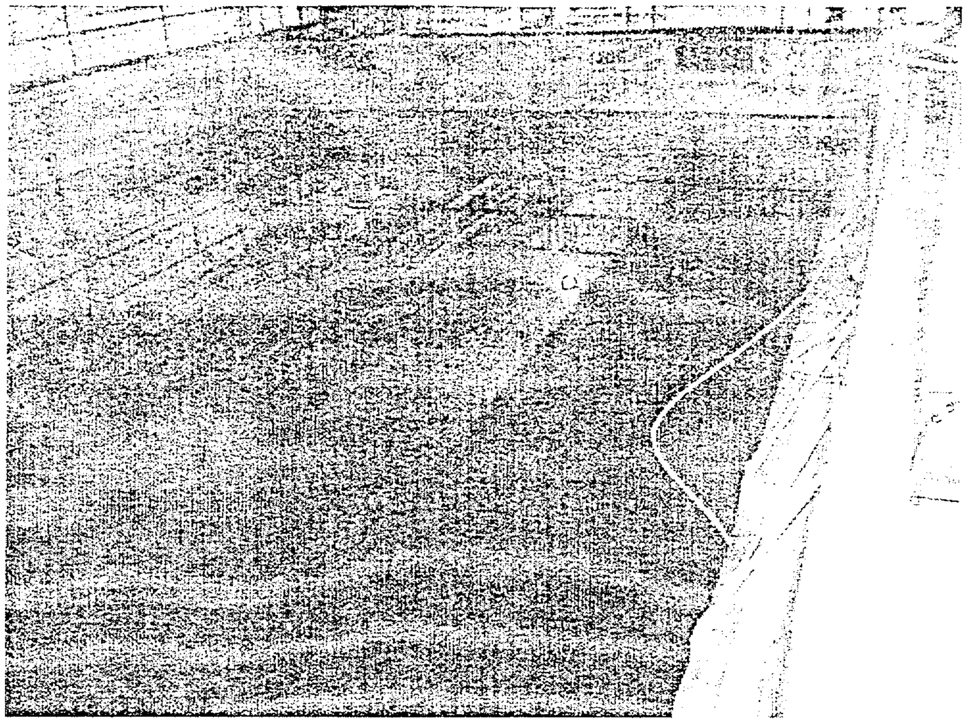


Figure F-29 – FHB El. 286'

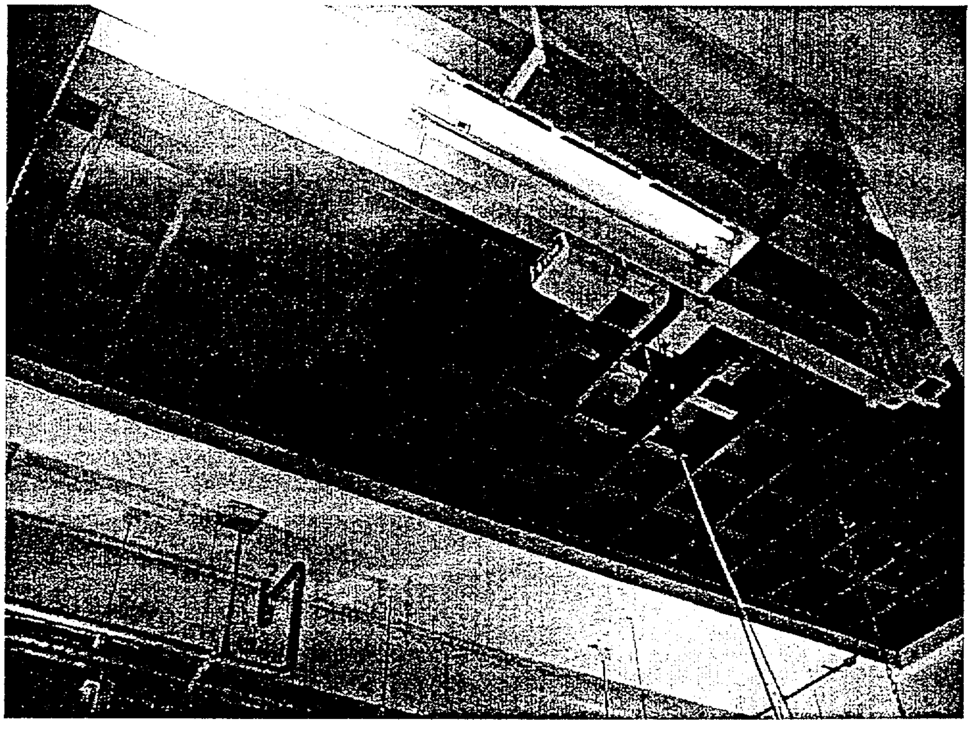


Figure F-30 – FHB El. 261' North Looking up at Eq Hatch

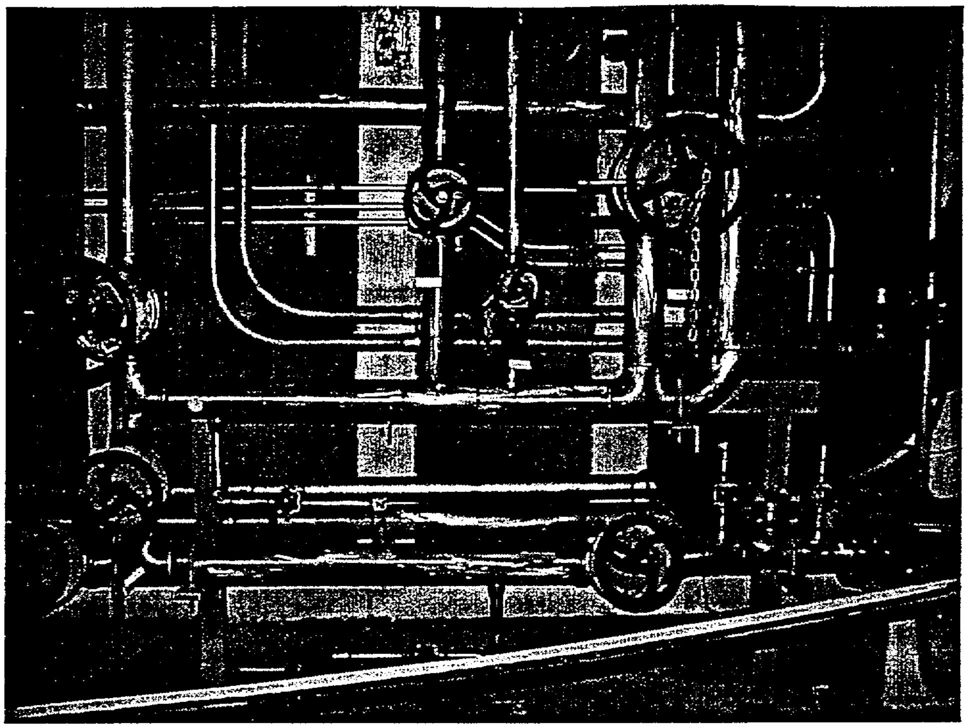


Figure F-31 – FHB El. 216' N, Fuel Pool Purification Pumps

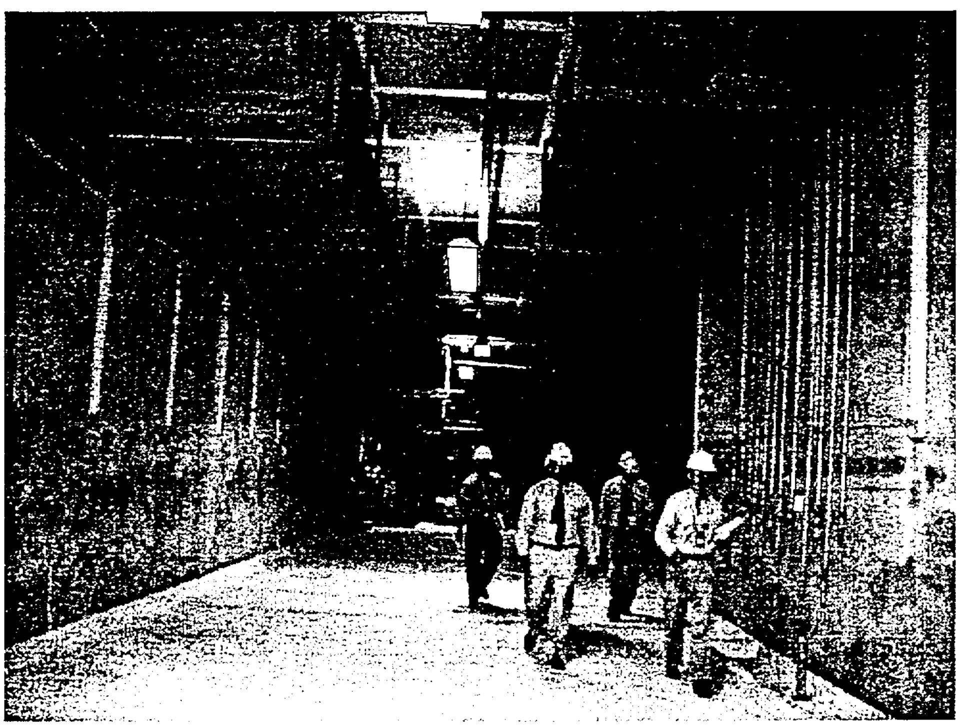


Figure F-32 – FHB El. 216' N Looking South

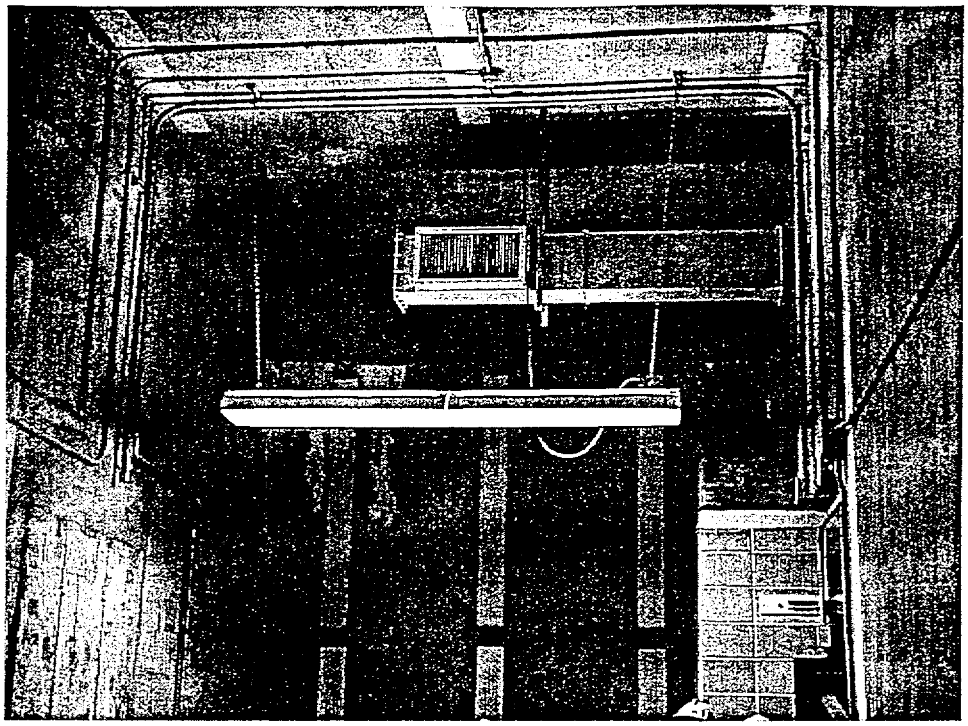


Figure F-33– FHB El. 216' N

Figure F-34 – FHB El. 236' N, Doorway to Safety Meeting Room

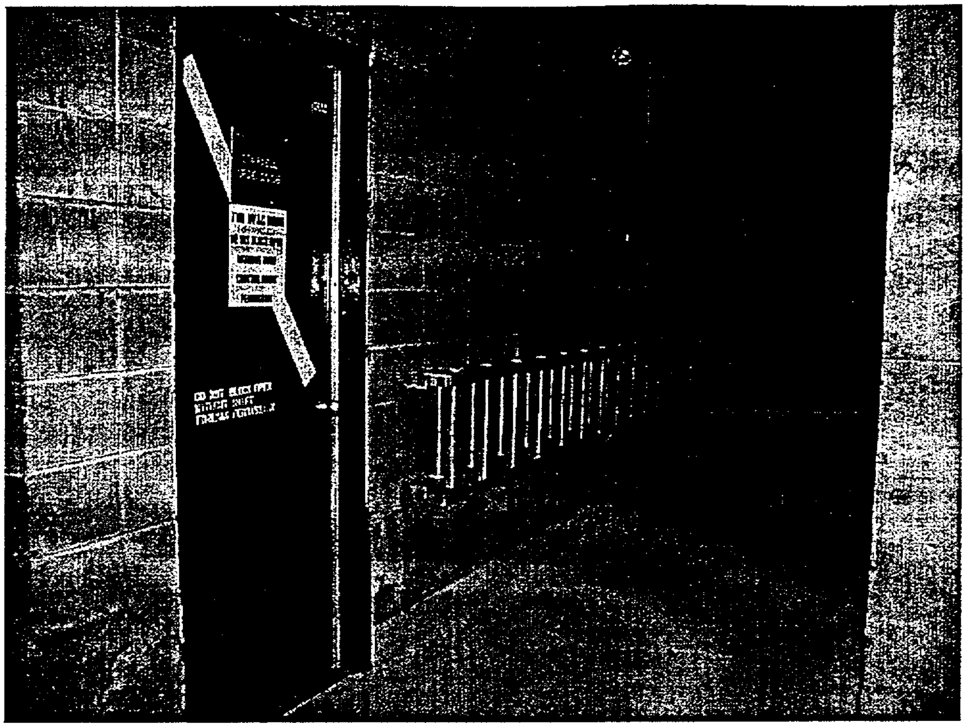


Figure F-35 – FHB El. 236' N, Stairwell Door and Ladder

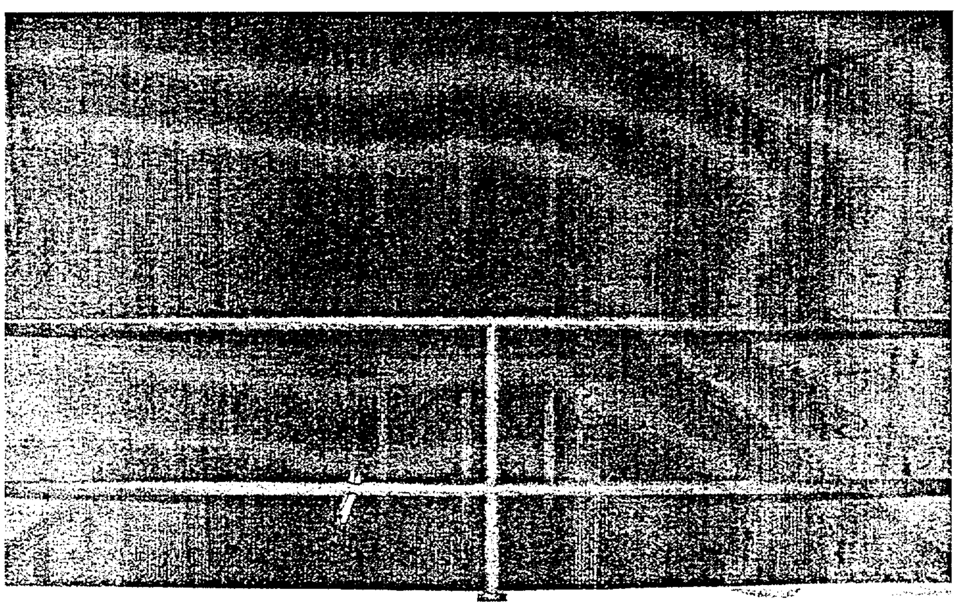


Figure F-36 – FHB El. 236' N, Opening to 216'

Figure F-37 – Simplified Fuel Handling Building Arrangement

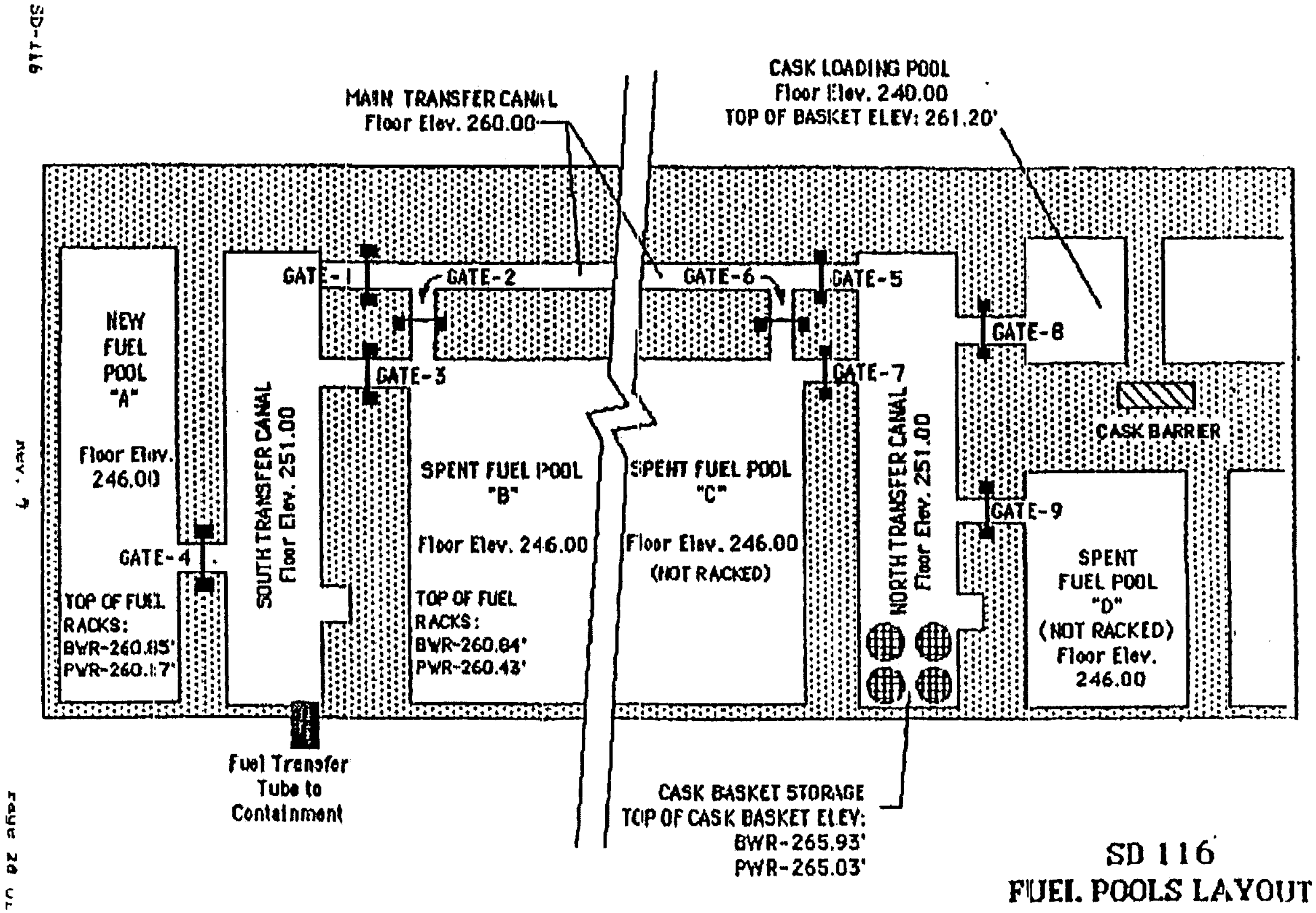
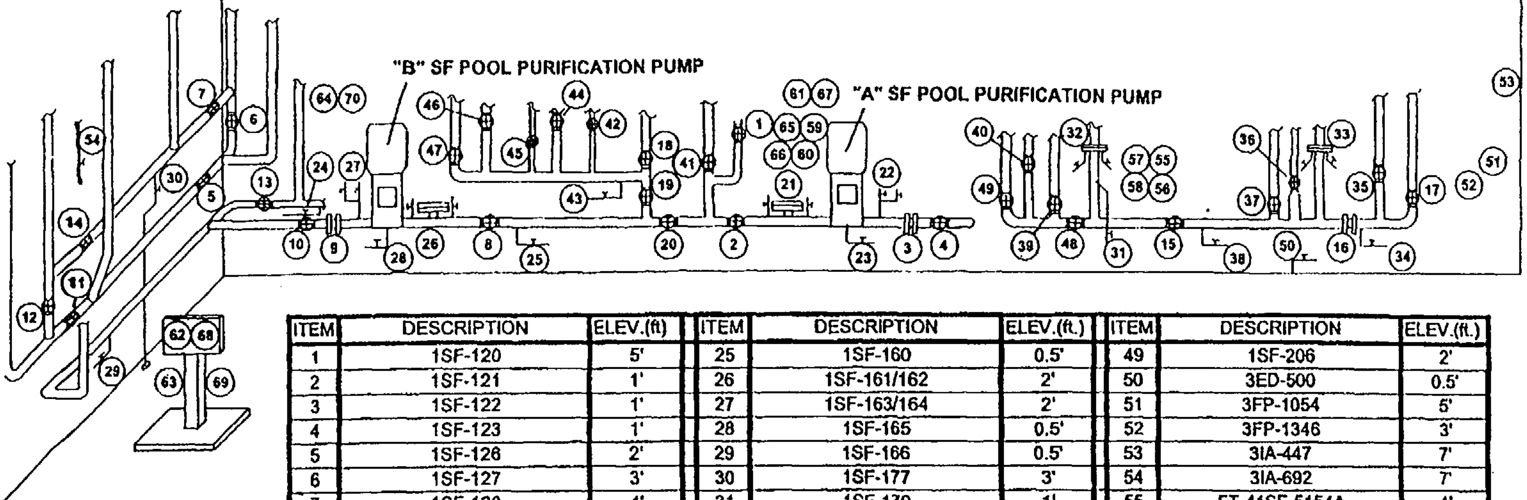


Figure F-38 – FHB 216' South – Spent Fuel Pool Purification

FHB 216' SOUTH - SPENT FUEL POOL PURIFICATION



ITEM	DESCRIPTION	ELEV.(ft)	ITEM	DESCRIPTION	ELEV.(ft.)	ITEM	DESCRIPTION	ELEV.(ft.)
1	1SF-120	5'	25	1SF-160	0.5'	49	1SF-206	2'
2	1SF-121	1'	26	1SF-161/162	2'	50	3ED-500	0.5'
3	1SF-122	1'	27	1SF-163/164	2'	51	3FP-1054	5'
4	1SF-123	1'	28	1SF-165	0.5'	52	3FP-1346	3'
5	1SF-126	2'	29	1SF-166	0.5'	53	3IA-447	7'
6	1SF-127	3'	30	1SF-177	3'	54	3IA-692	7'
7	1SF-130	4'	31	1SF-179	1'	55	FT-41SF-5154A	4'
8	1SF-131	1'	32	1SF-180/181	4'	56	FT-41SF-5154A-E/HD1/HD2/	1'-3'
9	1SF-132	1'	33	1SF-182/183	4'		HI1/HV1/LD1/LD2/LI1/LV1	
10	1SF-133	1'	34	1SF-184	0.5'	57	FT-41SF-5154B	4'
11	1SF-136	2'	35	1SF-187	3'	58	FT-41SF-5154B-E/HD1/HD2/	1'-3'
12	1SF-137	3'	36	1SF-188	3'		HI1/HV1/LD1/LD2/LI1/LV1	
13	1SF-138	1'	37	1SF-189	2'	59	PI-41SF-5190A	5'
14	1SF-139	4'	38	1SF-190	0.5'	60	PI-41SF-5190A-D1/D2/I1/V2	0.5'-4'
15	1SF-141	1'	39	1SF-191	2'	61	PI-41SF-5190A-V1	5'
16	1SF-142	1'	40	1SF-192	3'	62	PI-41SF-5190B	5'
17	1SF-143	2'	41	1SF-193	4'	63	PI-41SF-5190B-D1/D2/I1/V2	0.5'-3'
18	1SF-148	3'	42	1SF-194	4'	64	PI-41SF-5190B-V1	5'
19	1SF-149	2'	43	1SF-195	2'	65	PS-41SF-5190A	5'
20	1SF-150	1'	44	1SF-200	4'	66	PS-41SF-5190A-D1/D2/I1/V2	0.5'-4'
21	1SF-151/152	2'	45	1SF-201	3'	67	PS-41SF-5190A-V1	5'
22	1SF-153/154	2'	46	1SF-202	4'	68	PS-41SF-5190B	5'
23	1SF-155	0.5'	47	1SF-203	3'	69	PS-41SF-5190B-D1/D2/I1/V2	0.5'-3'
24	1SF-156	0.5'	48	1SF-205	1'	70	PS-41SF-5190B-V1	5'

Appendix G
SEISMIC ANALYSIS QUANTIFICATION DETAILS

The quantification of the seismic analysis was performed using Excel spreadsheet equations in place of event tree–fault tree codes. The spreadsheet equations include Boolean algebra where necessary. The spreadsheet calculational approach was employed to facilitate sensitivity calculations, and was practicable given the bounding nature of the analysis (e.g., loss of offsite power assumed, like component fragilities assumed completely dependent).

The overall quantification spreadsheets are provided here in Figures G-1 and G-2. Figure G-1 presents the calculation structure with seven seismic hazard ranges. Figure G-2 presents the calculation structure with sixteen seismic hazard ranges. The quantification process included additional worksheets in which certain key parameters of the process were quantified and documented (e.g., seismic hazard curve, fragility dependence curve); these other worksheets are not reproduced in this appendix.

The spreadsheet in Figure G-1 was used to perform the quantification of the Base Case and Sensitivity Cases 2 through 9. The spreadsheet in Figure G-2 was used to perform the quantification of Sensitivity Cases 1 through 10. Refer to Section 4.2 of this report for discussion of the results associated with these quantifications.

Figure G-1 SEISMIC QUANTIFICATION SPREADSHEET USING 7 SEISMIC MAGNITUDE RANGES

	<0.10 pga	0.1-0.3 pga	0.3-0.5 pga	0.5-0.7 pga	0.7-1.0 pga	1.0-1.5 pga	>1.5g
Seis. Range Frequency (1), (2)	n/a	1.93E-04	1.39E-05	2.89E-06	1.15E-06	3.01E-07	2.99E-07
Seis. Range Magnitude (1), (3)	n/a	0.2	0.4	0.6	0.85	1.25	1.5
Seismic-Induced LOOP Probability	n/a	1.0	1.0	1.0	1.0	1.0	1.0
EDG Non-Seismic CCF	n/a	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04
AC Recovery Failure Prob.	n/a	1.0	1.0	1.0	1.0	1.0	1.0
EDG Median Capacity (Am)	n/a	1.25	1.25	1.25	1.25	1.25	1.25
EDG Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57
EDG Fragility (3), (4)	n/a	5.99E-04	2.20E-02	9.72E-02	2.48E-01	5.00E-01	6.26E-01
Ess. SWGR Median Capacity (Am)	n/a	1.31	1.31	1.31	1.31	1.31	1.31
Ess. SWGR Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57
Ess. SWGR Fragility (3), (4)	n/a	4.46E-04	1.80E-02	8.37E-02	2.22E-01	4.67E-01	5.95E-01
Class IE Bldg. Median Capacity (Am)	n/a	1.50	1.50	1.50	1.50	1.50	1.50
Class IE Bldg. Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57
Class IE Bldg. Fragility (3), (4)	n/a	1.84E-04	9.73E-03	5.26E-02	1.58E-01	3.74E-01	5.00E-01
Containment Median Capacity (Am)	n/a	2.00	2.00	2.00	2.00	2.00	2.00
Containment Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57
Containment. Fragility (3), (4)	n/a	2.35E-05	2.22E-03	1.67E-02	6.52E-02	2.03E-01	3.06E-01
Seismic CDF (5)	negligible	2.60E-07	7.15E-07	6.97E-07	7.22E-07	3.72E-07	4.49E-07
Seismic CDF (Class IE & Cont. Bldgs)	n/a	4.00E-08	1.65E-07	1.98E-07	2.44E-07	1.51E-07	1.95E-07
Seismic CDF (w/o Class IE & Cont. Bldgs)	negligible	2.20E-07	5.50E-07	5.00E-07	4.77E-07	2.21E-07	2.54E-07
			Total Seismic CDF:				3.22E-06
			Total Seismic CDF (w/o Class IE & Cont. Bldgs):				2.22E-06
Seismic CDF (w/o Class IE & Cont. Bldgs)	negligible	2.20E-07	5.50E-07	5.00E-07	4.77E-07	2.21E-07	2.54E-07
Probability of Early Containment Failure	n/a	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	n/a
PCIV Median Capacity (Am) (7)	n/a	2.00	2.00	2.00	2.00	2.00	n/a
PCIV Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	n/a
PCIV Fragility (3), (4)	n/a	2.35E-05	2.22E-03	1.67E-02	6.52E-02	2.03E-01	n/a
PCIV Fragility Dependence	n/a	6.82E-02	1.03E-01	1.56E-01	2.61E-01	5.97E-01	n/a
Probability of Pre-Existing Containment Leakage	n/a	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	n/a
Probability of Non-Seismic Induced Isolation Failure	n/a	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	n/a
Probability of PCIV Manual Isolation Failure	n/a	1.0	1.0	1.0	1.0	1.0	n/a
Probability of Containment Isolation Failure	n/a	2.00E-03	2.23E-03	4.59E-03	1.90E-02	1.23E-01	n/a
Diesel Fire Pump Median Capacity (Am) (7)	n/a	1.25	1.25	1.25	1.25	1.25	n/a
Diesel Fire Pump Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	n/a
Diesel Fire Pump Fragility (3), (4)	n/a	5.99E-04	2.20E-02	9.72E-02	2.48E-01	5.00E-01	n/a
Diesel Fire Pump Failure to Run/Start	n/a	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	n/a
FHB Bldg. Flooding Median Capacity (Am) (7), (8)	n/a	1.25	1.25	1.25	1.25	1.25	n/a
FHB Bldg. Flooding Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	n/a
FHB Bldg. Flooding Fragility (3), (4)	n/a	5.99E-04	2.20E-02	9.72E-02	2.48E-01	5.00E-01	n/a
Conditional Probability Flood Prevents Access to Pool Deck	n/a	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	n/a
Conditional Probability Flood Prevents Access to Basement	n/a	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	n/a

pga	Shearon-Harris Annual Exceedance Frequency (NUREG-1488 curve-fit)
0.10	2.11E-04
0.20	5.10E-05
0.30	1.85E-05
0.40	8.89E-06
0.50	4.64E-06
0.60	2.69E-06
0.70	1.75E-06
0.80	1.14E-06
0.90	8.05E-07
1.00	6.00E-07
1.10	4.62E-07
1.20	4.08E-07
1.30	3.50E-07
1.40	3.09E-07
1.50	2.99E-07

Fragility Uncertainty & Randomness

BETA(u) =	0.40
BETA(r) =	0.40

Figure G-1 SEISMIC QUANTIFICATION SPREADSHEET USING 7 SEISMIC MAGNITUDE RANGES (cont.)

Fire Hose Alignment HEP (w/Late CF)	n/a	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	n/a
Fire Hose Alignment HEP (w/IS Failure)	n/a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	n/a
Fire Hose Alignment HEP (w/Early CF)	n/a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	n/a
Demin. Piping Alignment HEP (w/Late CF)	n/a	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	n/a
Demin. Piping Alignment HEP (w/IS Failure)	n/a	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	n/a
Demin. Piping Alignment HEP (w/Early CF)	n/a	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	n/a
Offsite Infrastructure Median Capacity (Am) (7)	n/a	1.00	1.00	1.00	1.00	1.00	n/a
Offsite Infrastructure Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	n/a
Offsite Infrastructure Fragility (3), (4)	n/a	2.22E-03	5.26E-02	1.83E-01	3.87E-01	6.53E-01	n/a
Infrastructure Failures Preclude Fire Truck Arrival at Site	n/a	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	n/a
Infrastructure Failures Preclude Portable Pump/Gen. Arrival at Si	n/a	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	n/a
Fire Truck Hook-Up HEP	n/a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	n/a
Portable pump/generator Hook-Up HEP	n/a	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	n/a
FHB Inventory Control Failure (w/Late CF) (9)	n/a	4.85E-03	7.41E-03	1.80E-02	4.59E-02	1.08E-01	n/a
FHB Inventory Control Failure (w/IS Failure) (9)	n/a	6.92E-02	8.85E-02	1.49E-01	2.59E-01	4.24E-01	n/a
FHB Inventory Control Failure (w/Early CF) (9)	n/a	1.46E-01	1.64E-01	2.19E-01	3.18E-01	4.64E-01	n/a
Seismic-Induced Spent Fuel Failure Frequency (10)	negligible	2.26E-09	7.40E-09	1.30E-08	2.87E-08	3.51E-08	n/a
		Total Seismic-Induced Spent Fuel Failure Frequency (with respect to ASLB scenari					8.65E-08

NOTES:

(1) Seismic hazard curve divided into 7 seismic ranges.
(2) Seismic range frequency is the annual frequency of a seismic event with magnitude within range (i.e., frequency of low end of range minus frequency of high end of range)
(3) Seismic range magnitude used in fragility calculations taken as the midpoint of the seismic range.
(4) Each SSC seismic fragility conservatively applies to all like SSCs (e.g., seismic induced failure of EDG means failure of all EDGs)
(5) Accident sequences comprising seismic CDF (total) calculation are:
- Seismic Event *Seismic-Induced LOOP * Seismic-Induced Failure of EDGs * AC Recovery Failure
- Seismic Event *Seismic-Induced LOOP * Non-Seismic CCFof EDGs * AC Recovery Failure
- Seismic Event *Seismic-Induced LOOP * Seismic-Induced Failure of Ess. SWGR * AC Recovery Failure
- Seismic Event *Seismic-Induced Class IE Bldg. Failures

Accident sequences involving seismic-induced containment/FHB failure are outside the scope of the analysis.
(6) BETAc = SQR(BETAr^2 + BETAu^2).
(7) Judgment.
(8) This item addresses potential seismic-induced failure of purification equipment and subsequent flooding, precluding access to key SFP Bldg. areas necessary for alignment of alternate SFP inventory control methods.
(9) Calculation of SFP inventory control failure calculated by summing the following scenarios:
- SFP Bldg. Access Precluded due to Seismic-Induced Flooding
- Failure of SFP alternate cooling alignment inside SFP Bldg. (no flooding)
- Success of SFP cooling alignment inside SFP Bldg. * DFP Failure * Failure of Other Pumping Sources (i.e., fire truck and portable pump/generator)

(10) The spent fuel pool seismic-induced loss of inventory does not include draindown events.

Figure G-1 SEISMIC QUANTIFICATION SPREADSHEET USING 7 SEISMIC MAGNITUDE RANGES (cont.)

BOOLEAN ADDITIONS/SUBSTRACTIONS FOR "FP" NODE:	<0.10 pga	0.1-0.3 pga	0.3-0.5 pga	0.5-0.7 pga	0.7-1.0 pga	1.0-1.5 pga
Cutsets For: "FHB Inventory Control Failure (w/Late CF)"	n/a	4.85E-03	7.43E-03	1.81E-02	4.65E-02	1.11E-01
First Batch of Boolean Intersection Cutsets	n/a	8.97E-08	4.60E-06	4.54E-05	2.98E-04	1.48E-03
Second Batch of Boolean Intersection Cutsets	n/a	4.31E-06	6.56E-06	1.64E-05	4.43E-05	1.11E-04
Third Batch of Boolean Intersection Cutsets	n/a	5.02E-08	1.71E-06	1.45E-05	8.45E-05	3.64E-04
Fourth Batch of Boolean Intersection Cutsets	n/a	1.69E-06	6.63E-06	2.99E-05	9.96E-05	2.69E-04
Fifth Batch of Boolean Intersection Cutsets	n/a	3.70E-09	3.22E-07	5.16E-06	3.71E-05	1.72E-04
Boolean Summation	n/a	4.85E-03	7.41E-03	1.80E-02	4.59E-02	1.08E-01
Cutsets For: "FHB Inventory Control Failure (w/IS Failure)"	n/a	7.02E-02	9.11E-02	1.58E-01	2.91E-01	5.17E-01
First Batch of Boolean Intersection Cutsets	n/a	2.09E-05	8.81E-04	5.34E-03	2.07E-02	6.68E-02
Second Batch of Boolean Intersection Cutsets	n/a	9.68E-04	1.16E-03	1.73E-03	2.81E-03	4.71E-03
Third Batch of Boolean Intersection Cutsets	n/a	1.13E-05	2.94E-04	1.33E-03	4.23E-03	1.19E-02
Fourth Batch of Boolean Intersection Cutsets	n/a	3.50E-05	2.92E-04	1.13E-03	2.97E-03	6.64E-03
Fifth Batch of Boolean Intersection Cutsets	n/a	6.54E-08	8.45E-06	1.23E-04	7.74E-04	3.27E-03
Boolean Summation	n/a	6.92E-02	8.85E-02	1.49E-01	2.59E-01	4.24E-01
Cutsets For: "FHB Inventory Control Failure (w/Early CF)"	n/a	1.51E-01	1.72E-01	2.39E-01	3.72E-01	5.98E-01
First Batch of Boolean Intersection Cutsets	n/a	4.52E-05	1.77E-03	9.28E-03	3.07E-02	8.70E-02
Second Batch of Boolean Intersection Cutsets	n/a	5.09E-03	6.11E-03	9.09E-03	1.48E-02	2.48E-02
Third Batch of Boolean Intersection Cutsets	n/a	1.13E-05	2.94E-04	1.33E-03	4.23E-03	1.19E-02
Fourth Batch of Boolean Intersection Cutsets	n/a	3.50E-05	2.92E-04	1.13E-03	2.97E-03	6.64E-03
Fifth Batch of Boolean Intersection Cutsets	n/a	6.54E-08	8.45E-06	1.23E-04	7.74E-04	3.27E-03
Boolean Summation	n/a	1.46E-01	1.64E-01	2.19E-01	3.18E-01	4.64E-01

Figure G-2 SEISMIC QUANTIFICATION SPREADSHEET USING 16 SEISMIC MAGNITUDE RANGES

	≤0.10 pga	0.1-0.2 pga	0.2-0.3 pga	0.3-0.4 pga	0.4-0.5 pga	0.5-0.6 pga	0.6-0.7 pga	0.7-0.8 pga	0.8-0.9 pga	0.9-1.0 pga	1.0-1.1 pga	1.1-1.2 pga	1.2-1.3 pga	1.3-1.4 pga	1.4-1.5 pga	>1.5 pga
Seis. Range Frequency (1), (2)	n/a	1.60E-04	3.25E-05	9.61E-06	4.25E-06	1.95E-06	9.40E-07	6.10E-07	3.35E-07	2.05E-07	1.18E-07	7.40E-08	5.80E-08	4.10E-08	1.00E-08	2.99E-07
Seis. Range Magnitude (1), (3)	n/a	0.15	0.25	0.35	0.45	0.55	0.65	0.75	0.85	0.95	1.05	1.15	1.25	1.35	1.45	1.5
Seismic-Induced LOOP Probability	n/a	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
EDG Non-Seismic CCF	n/a	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04	1.00E-04
AC Recovery Failure Prob.	n/a	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
EDG Median Capacity (Am)	n/a	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25
EDG Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57
EDG Fragility (3), (4)	n/a	8.91E-06	2.22E-03	1.22E-02	3.55E-02	7.33E-02	1.24E-01	1.83E-01	2.48E-01	3.14E-01	3.79E-01	4.41E-01	5.00E-01	5.54E-01	6.03E-01	6.26E-01
Ess. SWGR Median Capacity (Am)	n/a	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31	1.31
Ess. SWGR Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57
Ess. SWGR Fragility (3), (4)	n/a	6.38E-05	1.71E-03	9.82E-03	2.95E-02	6.25E-02	1.08E-01	1.62E-01	2.22E-01	2.85E-01	3.48E-01	4.09E-01	4.67E-01	5.21E-01	5.71E-01	5.95E-01
Class IE Bldg. Median Capacity (Am)	n/a	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Class IE Bldg. Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57
Class IE Bldg. Fragility, (3), (4)	n/a	2.35E-06	7.69E-04	5.06E-03	1.67E-02	3.81E-02	6.97E-02	1.10E-01	1.58E-01	2.10E-01	2.64E-01	3.19E-01	3.74E-01	4.26E-01	4.76E-01	6.00E-01
Containment Median Capacity (Am)	n/a	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00
Containment Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57
Containment. Fragility (3), (4)	n/a	2.34E-06	1.19E-04	1.03E-03	4.18E-03	1.12E-02	2.35E-02	4.15E-02	6.52E-02	9.41E-02	1.27E-01	1.64E-01	2.03E-01	2.44E-01	2.85E-01	3.06E-01
Seismic CDF (5)	negligible	4.46E-08	1.60E-07	2.70E-07	3.60E-07	3.51E-07	2.91E-07	2.82E-07	2.10E-07	1.63E-07	1.12E-07	8.15E-08	7.16E-08	5.54E-08	1.46E-08	4.49E-07
Seismic CDF (Class IE & Cont. Bldgs)	n/a	4.13E-09	2.88E-08	5.84E-08	8.83E-08	9.53E-08	8.60E-08	8.97E-08	7.12E-08	6.82E-08	4.22E-08	3.19E-08	2.90E-08	2.32E-08	6.25E-09	1.95E-07
Seismic CDF (w/o Class IE & Cont. Bldgs)	negligible	4.06E-08	1.31E-07	2.12E-07	2.72E-07	2.56E-07	2.05E-07	1.93E-07	1.39E-07	1.04E-07	7.02E-08	4.96E-08	4.25E-08	3.22E-08	8.30E-09	2.54E-07

Total Seismic CDF: 2.92E-06

Total Seismic CDF (w/o Class IE & Cont. Bldgs): 2.01E-06

	≤0.10 pga	0.1-0.2 pga	0.2-0.3 pga	0.3-0.4 pga	0.4-0.5 pga	0.5-0.6 pga	0.6-0.7 pga	0.7-0.8 pga	0.8-0.9 pga	0.9-1.0 pga	1.0-1.1 pga	1.1-1.2 pga	1.2-1.3 pga	1.3-1.4 pga	1.4-1.5 pga	>1.5 pga
Seismic CDF (w/o Class IE & Cont. Bldgs)	negligible	4.06E-08	1.31E-07	2.12E-07	2.72E-07	2.56E-07	2.06E-07	1.93E-07	1.39E-07	1.04E-07	7.02E-08	4.98E-08	4.25E-08	3.22E-08	8.30E-09	2.54E-07
Probability of Early Containment Failure	n/a	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	3.76E-02	n/a
PCIV Median Capacity (Am) (7)	n/a	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	2.00	n/a
PCIV Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	n/a
PCIV Fragility (3), (4)	n/a	2.34E-06	1.19E-04	1.03E-03	4.18E-03	1.12E-02	2.35E-02	4.15E-02	6.52E-02	9.41E-02	1.27E-01	1.64E-01	2.03E-01	2.44E-01	2.85E-01	n/a
PCIV Fragility Dependence	n/a	6.15E-02	7.56E-02	9.29E-02	1.14E-01	1.40E-01	1.73E-01	2.12E-01	2.61E-01	3.21E-01	3.95E-01	4.85E-01	5.97E-01	7.34E-01	9.02E-01	n/a
Probability of Pre-Existing Containment Leakage	n/a	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	n/a
Probability of Non-Seismic Induced Isolation Failure	n/a	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03	n/a
Probability of PCIV Manual Isolation Failure	n/a	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	n/a
Probability of Containment Isolation Failure	n/a	2.00E-03	2.01E-03	2.10E-03	2.48E-03	3.58E-03	6.06E-03	1.08E-02	1.90E-02	3.22E-02	5.23E-02	8.16E-02	1.23E-01	1.81E-01	2.69E-01	n/a
Diesel Fire Pump Median Capacity (Am) (7)	n/a	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	n/a
Diesel Fire Pump Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	n/a
Diesel Fire Pump Fragility (3), (4)	n/a	8.91E-05	2.22E-03	1.22E-02	3.55E-02	7.33E-02	1.24E-01	1.83E-01	2.48E-01	3.14E-01	3.79E-01	4.41E-01	5.00E-01	5.54E-01	6.03E-01	n/a
Diesel Fire Pump Failure to Run/Start	n/a	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	n/a
FHB Bldg. Flooding Median Capacity (Am) (7), (8)	n/a	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	n/a
FHB Bldg. Flooding Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	n/a
FHB Bldg. Flooding Fragility (3), (4)	n/a	8.91E-05	2.22E-03	1.22E-02	3.55E-02	7.33E-02	1.24E-01	1.83E-01	2.48E-01	3.14E-01	3.79E-01	4.41E-01	5.00E-01	5.64E-01	6.03E-01	n/a
Conditional Probability Flood Prevents Access to Pool Deck	n/a	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	n/a
Conditional Probability Flood Prevents Access to Basement	n/a	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	n/a
Fire Hose Alignment HEP (w/Late CF)	n/a	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	6.20E-02	n/a
Fire Hose Alignment HEP (w/S Failure)	n/a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	n/a
Fire Hose Alignment HEP (w/Early CF)	n/a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	n/a
Demin. Piping Alignment HEP (w/Late CF)	n/a	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	n/a
Demin. Piping Alignment HEP (w/S Failure)	n/a	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	1.90E-02	n/a
Demin. Piping Alignment HEP (w/Early CF)	n/a	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	n/a
Offsite Infrastructure Median Capacity (Am) (7)	n/a	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	n/a
Offsite Infrastructure Fragility BETAc (6)	n/a	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	0.57	n/a
Offsite Infrastructure Fragility (3), (4)	n/a	3.99E-04	7.13E-03	3.17E-02	7.90E-02	1.45E-01	2.23E-01	3.06E-01	3.87E-01	4.64E-01	5.34E-01	5.98E-01	6.53E-01	7.02E-01	7.44E-01	n/a
Infrastructure Failures Preclude Fire Truck Arrival at Site	n/a	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	n/a
Infrastructure Failures Preclude Portable Pump/Gen Arrival at Site	n/a	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01	n/a
Fire Truck Hook-Up HEP	n/a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	n/a
Portable pump/generator Hook-Up HEP	n/a	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	5.00E-02	n/a
FHB Inventory Control Failure (w/Late CF) (9)	n/a	4.78E-03	6.04E-03	6.22E-03	9.11E-03	1.43E-02	2.23E-02	3.29E-02	4.59E-02	6.06E-02	7.63E-02	9.24E-02	1.08E-01	1.24E-01	1.38E-01	n/a
FHB Inventory Control Failure (w/S Failure) (9)	n/a	6.86E-02	7.08E-02	8.00E-02	9.97E-02	1.30E-01	1.69E-01	2.13E-01	2.59E-01	3.05E-01	3.48E-01	3.88E-01	4.24E-01	4.56E-01	4.83E-01	n/a
FHB Inventory Control Failure (w/Early CF) (9)	n/a	1.46E-01	1.48E-01	1.56E-01	1.74E-01	2.02E-01	2.37E-01	2.77E-01	3.18E-01	3.59E-01	3.98E-01	4.33E-01	4.64E-01	4.92E-01	5.16E-01	n/a
Seismic-Induced Spent Fuel Failure Frequency (10)	negligible	4.13E-10	1.38E-09	2.54E-09	4.72E-09	5.58E-09	6.41E-09	8.47E-09	8.35E-09	8.29E-09	7.17E-09	6.36E-09	6.77E-09	6.29E-09	1.98E-09	n/a

Total Seismic-Induced Spent Fuel Failure Frequency (with respect to ASLB scenario): 7.42E-08

Shearon Harris Annual Exceedance Frequency (NUREG-1488 curve fit)

pga	
0.10	2.11E-04
0.20	5.10E-05
0.30	1.85E-05
0.40	8.89E-06
0.50	4.64E-06
0.60	2.69E-06
0.70	1.75E-06
0.80	1.14E-06
0.90	8.05E-07
1.00	6.00E-07
1.10	4.82E-07
1.20	4.08E-07
1.30	3.50E-07
1.40	3.09E-07
1.50	2.99E-07

Fragility Uncertainty & Randomness

BETA(u) = 0.40
BETA(r) = 0.40

Figure G-2 SEISMIC QUANTIFICATION SPREADSHEET USING 16 SEISMIC MAGNITUDE RANGES (cont.)

NOTES:

(1) Seismic hazard curve divided into 16 seismic ranges.
(2) Seismic range frequency is the annual frequency of a seismic event with magnitude within range (i.e., frequency of low end of range minus frequency of high end of range)
(3) Seismic range magnitude used in fragility calculations taken as the midpoint of the seismic range.
(4) Each SSC seismic fragility conservatively applies to all like SSCs (e.g., seismic induced failure of EDG means failure of all EDGs)
(5) Accident sequences comprising seismic CDF (total) calculation are:
- Seismic Event *Seismic-Induced LOOP * Seismic-Induced Failure of EDGs * AC Recovery Failure
- Seismic Event *Seismic-Induced LOOP * Non-Seismic CCF of EDGs * AC Recovery Failure
- Seismic Event *Seismic-Induced LOOP * Seismic-Induced Failure of Ess. SWGR * AC Recovery Failure
- Seismic Event *Seismic-Induced Class IE Bldg. Failures

Accident sequences involving seismic-induced containment/FHB failure are outside the scope of the analysis.
(6) BETAc = SQR(BETAr^2 + BETAu^2).
(7) Judgment
(8) This item addresses potential seismic-induced failure of purification equipment and subsequent flooding, precluding access to key SFP Bldg. areas necessary for alignment of alternate SFP inventory control methods.
(9) Calculation of SFP inventory control failure calculated by summing the following scenarios
- SFP Bldg. Access Precluded due to Seismic-Induced Flooding
- Failure of SFP alternate cooling alignment inside SFP Bldg. (no flooding)
- Success of SFP cooling alignment inside SFP Bldg. * DFP Failure * Failure of Other Pumping Sources (i.e., fire truck and portable pump/generator)

(10) The spent fuel pool seismic-induced loss of inventory does not include draindown events.

BOOLEAN ADDITIONS/SUBSTRACTIONS FOR "FP" NODE:

	≤0.10 pga	0.1-0.2 pga	0.2-0.3 pga	0.3-0.4 pga	0.4-0.5 pga	0.5-0.6 pga	0.6-0.7 pga	0.7-0.8 pga	0.8-0.9 pga	0.9-1.0 pga	1.0-1.1 pga	1.1-1.2 pga	1.2-1.3 pga	1.3-1.4 pga	1.4-1.5 pga
Cutsets For: "FHB Inventory Control Failure (w/Late CF)"	n/a	4.79E-03	5.05E-03	6.24E-03	9.14E-03	1.44E-02	2.25E-02	3.33E-02	4.65E-02	6.15E-02	7.76E-02	9.42E-02	1.11E-01	1.27E-01	1.42E-01
First Batch of Boolean Intersection Cutsets	n/a	1.32E-08	3.43E-07	2.22E-06	8.83E-06	2.76E-05	7.15E-05	1.57E-04	2.98E-04	5.04E-04	7.74E-04	1.10E-03	1.48E-03	1.88E-03	2.30E-03
Second Batch of Boolean Intersection Cutsets	n/a	4.25E-06	4.48E-06	5.51E-06	8.08E-06	1.29E-05	2.05E-05	3.11E-05	4.43E-05	5.96E-05	7.62E-05	9.35E-05	1.11E-04	1.28E-04	1.43E-04
Third Batch of Boolean Intersection Cutsets	n/a	8.92E-09	1.66E-07	8.82E-07	3.12E-06	9.06E-06	2.22E-05	4.64E-05	8.45E-05	1.37E-04	2.03E-04	2.80E-04	3.64E-04	4.52E-04	5.41E-04
Fourth Batch of Boolean Intersection Cutsets	n/a	1.57E-06	2.04E-06	4.28E-06	1.01E-05	2.16E-05	4.02E-05	6.64E-05	9.96E-05	1.38E-04	1.81E-04	2.25E-04	2.69E-04	3.12E-04	3.54E-04
Fifth Batch of Boolean Intersection Cutsets	n/a	6.25E-10	1.41E-08	1.26E-07	7.41E-07	2.93E-06	8.49E-06	1.94E-05	3.71E-05	6.21E-05	9.38E-05	1.31E-04	1.72E-04	2.15E-04	2.58E-04
Boolean Summation	n/a	4.78E-03	5.04E-03	6.22E-03	9.11E-03	1.43E-02	2.23E-02	3.29E-02	4.59E-02	6.06E-02	7.63E-02	9.24E-02	1.08E-01	1.24E-01	1.38E-01
Cutsets For: "FHB Inventory Control Failure (w/IS Failure)"	n/a	6.96E-02	7.20E-02	8.19E-02	1.03E-01	1.37E-01	1.82E-01	2.34E-01	2.91E-01	3.50E-01	4.08E-01	4.64E-01	5.17E-01	5.66E-01	6.12E-01
First Batch of Boolean Intersection Cutsets	n/a	3.10E-06	7.87E-05	4.63E-04	1.52E-03	3.69E-03	7.43E-03	1.31E-02	2.07E-02	3.02E-02	4.14E-02	5.37E-02	6.68E-02	8.02E-02	9.35E-02
Second Batch of Boolean Intersection Cutsets	n/a	9.61E-04	9.86E-04	1.08E-03	1.27E-03	1.55E-03	1.92E-03	2.35E-03	2.81E-03	3.30E-03	3.79E-03	4.26E-03	4.71E-03	5.14E-03	5.53E-03
Third Batch of Boolean Intersection Cutsets	n/a	2.02E-06	3.65E-05	1.70E-04	4.65E-04	9.76E-04	1.76E-03	2.84E-03	4.23E-03	5.90E-03	7.80E-03	9.84E-03	1.19E-02	1.41E-02	1.61E-02
Fourth Batch of Boolean Intersection Cutsets	n/a.	2.68E-05	5.80E-05	1.81E-04	4.41E-04	8.59E-04	1.43E-03	2.15E-03	2.97E-03	3.86E-03	4.79E-03	5.73E-03	6.64E-03	7.51E-03	8.32E-03
Fifth Batch of Boolean Intersection Cutsets	n/a	1.03E-08	2.87E-07	3.22E-06	1.93E-05	7.21E-05	1.95E-04	4.22E-04	7.74E-04	1.25E-03	1.85E-03	2.53E-03	3.27E-03	4.04E-03	4.81E-03
Boolean Summation	n/a	6.86E-02	7.08E-02	8.00E-02	9.97E-02	1.30E-01	1.69E-01	2.13E-01	2.59E-01	3.05E-01	3.48E-01	3.88E-01	4.24E-01	4.56E-01	4.83E-01
Cutsets For: "FHB Inventory Control Failure (w/Early CF)"	n/a	1.51E-01	1.53E-01	1.63E-01	1.84E-01	2.18E-01	2.63E-01	3.15E-01	3.72E-01	4.31E-01	4.89E-01	5.45E-01	5.98E-01	6.47E-01	6.93E-01
First Batch of Boolean Intersection Cutsets	n/a	6.71E-06	1.69E-04	9.58E-04	2.96E-03	6.66E-03	1.24E-02	2.06E-02	3.07E-02	4.29E-02	5.67E-02	7.16E-02	8.70E-02	1.03E-01	1.18E-01
Second Batch of Boolean Intersection Cutsets	n/a	5.06E-03	5.19E-03	5.68E-03	6.67E-03	8.17E-03	1.01E-02	1.23E-02	1.48E-02	1.74E-02	1.99E-02	2.24E-02	2.48E-02	2.70E-02	2.91E-02
Third Batch of Boolean Intersection Cutsets	n/a	2.02E-06	3.65E-05	1.70E-04	4.65E-04	9.76E-04	1.76E-03	2.84E-03	4.23E-03	5.90E-03	7.80E-03	9.84E-03	1.19E-02	1.41E-02	1.61E-02
Fourth Batch of Boolean Intersection Cutsets	n/a	2.68E-05	5.80E-05	1.81E-04	4.41E-04	8.59E-04	1.43E-03	2.15E-03	2.97E-03	3.86E-03	4.79E-03	5.73E-03	6.64E-03	7.51E-03	8.32E-03
Fifth Batch of Boolean Intersection Cutsets	n/a	1.03E-08	2.87E-07	3.22E-06	1.93E-05	7.21E-05	1.95E-04	4.22E-04	7.74E-04	1.25E-03	1.85E-03	2.53E-03	3.27E-03	4.04E-03	4.81E-03
Boolean Summation	n/a	1.46E-01	1.48E-01	1.56E-01	1.74E-01	2.02E-01	2.37E-01	2.77E-01	3.18E-01	3.59E-01	3.98E-01	4.33E-01	4.64E-01	4.92E-01	5.16E-01