Lewis Sumner
Vice President
Hatch Project Support

Southern Nuclear Operating Company, Inc.
40 Inverness Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279
Fax 205.992.0341

SOUTHERN COMPANY
Energy to Serve Your World™

July 6, 1998

Docket Nos. 50-321
50-366

HL-5647

Tac Nos. M99393
M99394

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

Edwin I. Hatch Nuclear Plant
Request for Additional Information:
Extended Power Uprate License Amendment Request

Gentlemen:

By letter dated March 27, 1998, the Nuclear Regulatory Commission (NRC) staff requested additional information regarding the Southern Nuclear Operating Company (SNC) Extended Power Uprate License Amendment Request submitted on August 8, 1997. By letter dated May 6, 1998, SNC provided responses to the questions included in the March 27th request with the exception of NRC Questions 56, 59, and 60. Enclosure 1 provides SNC's responses to NRC Questions 56, 59, and 60.

To support the SNC responses in Enclosure 1, General Electric (GE) prepared the document "SHEX Model Description" provided in Enclosure 2. This document contains proprietary information; therefore, the GE affidavit is provided in Enclosure 3.

Subsequent to the NRC's March 27th request, the NRC staff reviewers requested additional information regarding the subject amendment request through various forms of communication coordinated by the NRC Plant Hatch Project Manager. Enclosure 4 provides SNC's responses to these requests.

Should you have any questions in this regard, please contact this office.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

APOI 1/1

Change: NRC PDR 1 encl 1 non prop
Drawings located in Central File

Docket Nos. 50-321 HL-5647
50-366

Tac Nos. M99393
M99394

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

Edwin I. Hatch Nuclear Plant
Request for Additional Information:
Extended Power Uprate License Amendment Request

Gentlemen:

By letter dated March 27, 1998, the Nuclear Regulatory Commission (NRC) staff requested additional information regarding the Southern Nuclear Operating Company (SNC) Extended Power Uprate License Amendment Request submitted on August 8, 1997. By letter dated May 6, 1998, SNC provided responses to the questions included in the March 27th request with the exception of NRC Questions 56, 59, and 60. Enclosure 1 provides SNC's responses to NRC Questions 56, 59, and 60.

To support the SNC responses in Enclosure 1, General Electric (GE) prepared the document "SHEX Model Description" provided in Enclosure 2. This document contains proprietary information; therefore, the GE affidavit is provided in Enclosure 3.

Subsequent to the NRC's March 27th request, the NRC staff reviewers requested additional information regarding the subject amendment request through various forms of communication coordinated by the NRC Plant Hatch Project Manager. Enclosure 4 provides SNC's responses to these requests.

Should you have any questions in this regard, please contact this office.

Sincerely,

H. L. Sumner, Jr.

Enclosures:

1. SNC Response to NRC Questions 56, 59, and 60
2. SHEX Model Description
3. General Electric Company Affidavit
4. SNC Response to Additional NRC Questions
5. SCS Calculations SMNH-97-007 and SMNH-97-008
6. Drawing E-10173, Revision 7
7. Primary Meteorological Tower and Plant Structures
8. Page Change to Licensing Submittal

TWL/eb

cc: *Southern Nuclear Operating Company*
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request for Additional Information:
Extended Power Uprate License Amendment Request

SNC Response to NRC Questions 56, 59, and 60

NRC QUESTION 56

Enclosure 6 of NEDC-32749P, Section 4.1, did not include confirmatory calculations with the SHEX code and the HXSIZ code at the extended power level. Please provide the comparative analysis results. Similar decay heat models should be used in SHEX code for both confirmatory and extended power level analyses for results to remain comparable.

SNC RESPONSE

Background

The containment analyses for the Southern Nuclear Operating Company (SNC) extended power uprate submittal, dated August 8, 1997, were performed such that the results of one calculation bounded both Unit 1 and Unit 2. Due to a staff concern with the decay heat model input assumptions provided in the bounding analyses, confirmatory analyses with respect to the original power level (2436 MWt) were performed. The Final Safety Analysis Report (FSAR) analyses discussed in this response are the analyses performed assuming the original power level of 2436 MWt, as documented in revision 15A submitted on March 26, 1997. In a meeting with NRC staff on June 4, 1998, SNC proposed the performance of a unit specific approach including ANS 5.1 + 2 σ decay heat inputs to respond to this NRC request. Also, SNC presented preliminary benchmarking calculation results.

The reviewed and verified calculations included in this response follow the unit-specific approach discussed on June 4. However, the results have been revised from the preliminary results presented in the meeting. The revision is due to inconsistent peak wetwell airspace pressure values resulting from different input assumptions regarding break flow mixing with the drywell airspace and spray mixing with the drywell and wetwell airspace.

The HXSIZ code is not the code of record used for the original long-term containment response analyses provided in the FSAR. The FSAR analyses used a predecessor to the HXSIZ code, and the predecessor code is no longer available for performing confirmatory calculations.

General Electric currently uses the SHEX code as the model for BWR containment analyses, and SHEX was used for the extended power uprate analyses. SHEX models the vessel, drywell airspace, drywell pools, wetwell airspace, suppression pool, and system flows. Mass and energy balances are performed for each region. A mechanistic spray heat transfer model is used for the wetwell and drywell. A detailed model description of the SHEX code is provided in Enclosure 2. The long-term containment analysis is quasi-steady state. The period of interest is well after the vessel blowdown and vent clearing period when rapid changes are occurring.

A major difference in the long-term containment response calculation using the FSAR method versus the SHEX method is that the FSAR calculation assumed the drywell and wetwell airspace temperatures were equal to the spray temperature from the start of containment sprays. SHEX mechanistically models spray heat transfer, and thus, the results reflect the time required to bring the airspace temperature to spray temperature.

The following discussion provides a comparison between the SHEX code calculations and the original FSAR results for the long-term suppression pool temperature and wetwell pressure resulting from a DBA LOCA. A comparison between the containment pressure and temperature responses at the FSAR power level and at the extended power uprate power level is also provided to show the effect of the reactor power increase.

Approach

A calculation of the long-term containment response with SHEX was performed with the FSAR inputs. This benchmark analysis was compared to the containment pressure and pool temperature curves documented in FSAR. The comparison was done at FSAR conditions (105% steam flow and 104% rated thermal power).

The calculated containment pressure and temperature response for the FSAR power level is compared to analysis results for the extended power uprate power level to show the effect due to the requested power increase. Inputs for the calculation used for this comparison reflect the current plant geometry and are taken from extended power uprate analyses. Relative to the analysis discussed in NEDC-32749P, minor geometry input updates were made for theses analyses so that the calculation results are specific to Unit 2. In addition, the updated analyses use the Technical Specifications value for RHR flow rate of 7700 gpm and use 1979 ANS 5.1 + 2σ decay heat inputs.

Comparison of SHEX with FSAR

To validate the use of SHEX for Plant Hatch containment analyses, a benchmark analysis was performed with input assumptions consistent with the inputs used in an original FSAR analysis for Plant Hatch Unit 2 as discussed below.

Plant Input Conditions

Plant Hatch Unit 2 was chosen for the benchmark comparison because the Unit 2 FSAR included more detailed information than the Unit 1 FSAR with regard to the geometry inputs, power level, and systems operation. The inputs to the SHEX code were taken directly from the FSAR where possible. Inputs to the benchmark analysis were taken from the extended power uprate analyses if not documented in the FSAR.

Plant Configuration Assumptions

The long-term analysis assumes a double-ended recirculation suction line break with no offsite power and the assumed failure of one diesel generator. For this case, there is one division of the residual heat removal (RHR) system available for long-term containment cooling consisting of one heat exchanger, one pump and two RHR service water (RHRSW) pumps. One loop of the core spray (CS) system is also available. This containment cooling configuration is the limiting configuration with respect to the maximum suppression pool temperature. The benchmark analysis assumed the following ECCS and containment cooling configuration.

1. With a signal for LPCI initiation, one RHR pump automatically starts in the vessel-injection mode for the first 10 min of the event, when operator actions are not credited. The rated RHR pump flow of 7700 gpm per pump is assumed.

2. After receiving a signal for CS initiation, the one CS pump is injecting into the vessel. The CS pump flow is assumed to be 4625 gpm per pump.

3. At 10 min into the event, it is assumed the operator initiates containment cooling. The operator shuts off RHR injection to the vessel and aligns the RHR loop with two RHRSW pumps and one RHR heat exchanger for containment cooling using drywell and wetwell sprays. The rated RHR pump flow rate of 7700 gpm is assumed.

Other Input Assumptions

Input assumptions are used which match as closely as possible the input assumptions used in the FSAR analysis. These inputs are described below.

1. The reactor is assumed to be operating at 2537 MWt and 1025 psia initially, which corresponds to 104% of rated power and 105% of rated steam flow.

2. Vessel blowdown flow rates are based on the Homogeneous Equilibrium Model (Reference 1).

3. The core decay heat is a conservatively calculated May-Witt decay heat curve, which is listed in the FSAR.

4. Feedwater flow into the RPV is assumed to stop at time zero.

5. Thermodynamic equilibrium exists between the liquids and gases in the drywell. Mechanistic heat and mass transfer between the suppression pool and the suppression chamber airspace are modeled by SHEX. However, the evaporation rates from the pool surface to the wetwell airspace was set to near zero. These assumptions result in wetwell temperatures which approximate the FSAR assumption that the wetwell temperature is equal to the spray temperature.

6. No heat transfer from break fluids to the drywell atmosphere is assumed. This assumption is similar to the FSAR assumption that the drywell temperature is equal to the spray temperature.

7. The vent system flow to the suppression pool consists of a homogeneous mixture of the fluid in the drywell.

8. The initial suppression pool volume is at the low water level for Unit 2 (86,420 ft^3).

9. The initial drywell and suppression chamber airspace pressures are 0.75 psig.

10. An initial bulk average drywell temperature of 135°F and a relative humidity of 20% are used.

11. The initial suppression pool temperature is 95°F.

12. The initial suppression chamber airspace temperature is at 95°F, and the initial relative humidity is 100%.

13. The RHRSW temperature is the maximum allowable value of 95°F.

14. Drywell and wetwell heat sinks are not modeled in the containment analysis.

15. The RHR cooling system pumps have 100% of their horsepower rating converted to a pump heat input which is added either to the RPV liquid or suppression pool water.

16. Heat transfer from the primary containment to the reactor building is conservatively neglected.

17. Containment leakage is not included in the calculation of containment pressure response. Including the Technical Specifications containment leakage value of 1.2% per day will have no impact on the peak suppression pool temperature, but will slightly reduce the calculated containment pressure.

18. Drywell and suppression chamber sprays operate with 100% thermal mixing efficiency between the spray liquid and the drywell and suppression chamber atmosphere. This assumption is used to approximate the FSAR assumption that the drywell and wetwell temperatures are equal to the spray temperature.

Results

The reasonable agreement between the long-term (> 600 sec after accident) suppression pool temperature and containment pressure responses from the SHEX benchmark and the FSAR analysis show that the methodologies are essentially equivalent given the same inputs. The key results from the FSAR and the SHEX analysis are summarized in Table 56-1.

Upon examination of the calculations and results in the FSAR analysis, it was evident that the drywell and wetwell temperatures were assumed to equal the spray temperature at the onset of the sprays. This assumption results in an unrealistically minimized containment pressure response, especially near the onset of sprays. Therefore, the validation process is only intended to demonstrate that the SHEX code produces results consistent with those of the FSAR methodology given the same input assumptions, and that any differences between the results can be explained by the differences between drywell and wetwell temperature assumptions.

The maximum suppression pool temperature calculated by SHEX agrees well with the FSAR calculated value as shown in Table 56-1. Figure 56-1 provides a comparison of the suppression pool time history calculated using the FSAR methods with the benchmark analysis results. While there is a small inconsistency (~ 2°F) between the original FSAR text and figure documenting peak suppression pool temperature, Figure 56-1 shows that the shape of the long-term suppression pool temperature curve from the SHEX benchmark analysis matches well with the corresponding curve (Case c) reported in the FSAR. It should be noted that the curves given in Figures 56-1 and 56-2 are taken from an historical revision of the FSAR and do not represent the current peak suppression pool temperature documented in the FSAR.

Figure 56-2 compares the calculated wetwell pressure from the SHEX benchmark case with the FSAR wetwell pressure curves. The maximum long-term wetwell pressure (after 600 sec) calculated by SHEX using FSAR input assumptions is less than 0.5 psi different from the FSAR value (Case c in Figure 56-2). In addition, the shape of the containment pressure curve from 10,000 sec to past the time of the peak suppression pool temperature agrees well with the corresponding FSAR curve.

The containment pressure response from the onset of sprays (600 sec) to 10,000 sec is 1-3 psi higher than the values reported in the FSAR. This is due to the assumption used in the FSAR analysis that the drywell and wetwell temperatures are instantaneously equal to the spray temperature from the onset of sprays. This assumption neglects mechanistic heat transfer from the drywell to the spray droplets which is modeled by SHEX.

Comparison to Hand Calculations

To confirm the FSAR assumption that the drywell and wetwell temperature is set equal to the spray temperature, alternate calculations were performed using mass and energy presented in balances and classical thermodynamic relationships. The calculations confirm that the stated assumption in the FSAR regarding drywell and wetwell temperatures with spray operation is consistent with the FSAR analysis. The calculations also demonstrate the wetwell pressure of 12.4 psig determined from the SHEX benchmark analysis near the time of the peak suppression pool temperature is consistent with calculations based on classical thermodynamic relationships.

The drywell temperature shown for Case c at 600 sec is approximately 135°F, as shown in Figure 56-3. This temperature can be compared to the spray temperature to determine if the assumption regarding drywell temperature is true.

Calculation

The spray temperature can be estimated as the pool temperature minus the heat removed in the heat exchangers, plus the heat added to the fluid by the RHR pumps. The expression for the spray temperature is:

$$T_{spray}(600s) = T_{sup\,pool} + \frac{-k\left(T_{sup\,pool} + \frac{Q_{RHRpump}}{\dot{m}_{RHR}\cdot c} - T_{sw}\right)}{\dot{m}_{RHR}\cdot c} + \frac{Q_{RHRpump}}{\dot{m}_{RHR}\cdot c}$$

where

$T_{sup\,pool}(\approx 600s) = 150°\,F$

$k = 213.9\ Btu/s{\cdot}°F$, *heat exchanger K value*

$T_{sw} = 95°\,F$, *service water temperature*

$Q_{RHRpump} = 1501\ hp = 1060.7\ Btu/s$, *RHR pump heat*

c = specific heat of water = $1\ Btu/lb°\,F$

$\dot{m}_{RHR} = 7700\,gpm = 1070.3\ lbm/s$

$$T_{spray} = 150 + \frac{213.9\left(150 + \frac{1060.7}{1070.3} - 95\right)}{(1)(1070.3)} + \frac{1060.7}{1070.3} = 139.8°\,F$$

The suppression pool temperature at 600 sec is the value calculated by the FSAR analysis. The remaining parameters are the input parameters used in the FSAR analysis.

The results of these hand calculations show that for Case c the drywell temperature at 600 sec matches well with the spray temperature. This comparison confirms that the drywell temperature was assumed to be equal to the spray temperature in the FSAR methodology, as stated by Assumption B on Unit 2 FSAR page 6.2-53g.

The following hand calculation is used to estimate the wetwell pressure at the time of the peak suppression pool temperature. Near the time of the peak suppression pool temperature, quasi-equilibrium conditions exist in the wetwell and drywell, and the following input conditions exist:

$$T_{\sup pool}(\approx 32000s) = 210.1° F$$
$$k = 213.9\ Btu / s\cdot° F,\ heat\ exchanger\ K\ value$$
$$T_{sw} = 95° F,\ service\ water\ temperature$$
$$Q_{RHRpump} = 1501\ hp = 1060.7\ Btu / s,\ RHR\ pump\ heat$$
$$\dot{m}_{spray,dw} = 7300\ gpm = 1014.7\ lbm / s$$
$$\dot{m}_{spray,ww} = 400\ gpm = 55.6\ lbm / s$$
$$Q_{decay}(\approx 32000s) = decay\ heat \times initial\ power = 0.0095 \times 2.4046 \times 10^6 = 22843.7\ Btu / s$$
$$\Delta P_{vb} = 0.5,\ pressure\ difference\ across\ vacuum\ breakers$$
$$V_{dw} \cong initial\ V_{dw} = 1.46 \times 10^5\ ft^3$$
$$R = 53.34\ ft \cdot lbf / lbm° R,\ gas\ constant\ for\ air$$

$$T_{spray}(32000s) = T_{\sup pool} + \frac{-k\left(T_{\sup pool} + \frac{Q_{RHRpump}}{\dot{m}_{RHR}\cdot c} - T_{sw}\right)}{\dot{m}_{RHR}\cdot c} + \frac{Q_{RHRpump}}{\dot{m}_{RHR}\cdot c} =$$
$$210.1 + \frac{-213.9\left(210.1 + \frac{1060.7}{1070.3} - 95\right)}{1070.3} + \frac{1060.7}{1070.3} = 187.9° F$$

The wetwell and drywell temperatures are assumed equal to the spray temperature as in the FSAR analysis. The drywell and wetwell spray flow rates are found in the process diagram (Reference 2).

The total mass of the air in the drywell and wetwell are assumed to be constant. Therefore, the total air mass is equal to the initial air masses in the drywell and wetwell, which can be calculated to a reasonable approximation with the ideal gas law and the air-water vapor mixture relations.

$$initial\ M_{a,dw} = \frac{(initial\ P_{dw} - initial\ vapor\ P_{dw})initial\ V_{dw}}{R\cdot initial\ T_{dw}} = \frac{(15.45 - 0.2\cdot 2.5)(1.46\times 10^5)(144)}{53.34\cdot(135+460)} = 9903$$
$$initial\ M_{a,ww} = \frac{(initial\ P_{ww} - initial\ vapor\ P_{ww})initial\ V_{ww}}{R\cdot initial\ T_{ww}} = \frac{(15.45 - .82)(1.098\times 10^5)(144)}{53.34\cdot(95+460)} = 7814$$
$$M_{a,tot} = 9903 + 7814 = 17717$$

The volume of air in the wetwell at the time of maximum suppression pool temperature can be estimated as the initial wetwell volume minus the liquid volume lost through the break flow, and is calculated to be about 102,000 ft^3. The reduction in wetwell airspace volume is not large, and will not influence the results to a large extent.

If the drywell temperature is assumed to equal the spray temperature at the time of peak pool temperature, the corresponding containment pressure can be estimated with the following hand calculations.

$$P_{sat}(T_{dw}) = P_{sat}(T_{spray}) = 8.9\,psia$$

$$P_{sat}(T_{ww}) = P_{sat}(T_{spray}) = 8.9\,psia$$

$$M_{a,ww} = \frac{\left[\left(P_{sat}(T_{dw}) - P_{sat}(T_{ww}) + \Delta P_{vb}\right)\right] + \left[M_{a,tot}R(T_{dw}+460)/V_{dw}\right]}{\frac{R}{V_{dw}}(T_{dw}+460) + \frac{R}{V_{ww}}(T_{ww}+460)} =$$

$$\frac{[8.9 - 8.9 + 5]144 + \left[17717 \cdot 53.34(187.9+460)/1.46\times10^5\right]}{\frac{53.34}{1.46\times10^5}(187.9+460) + \frac{53.34}{1.021\times10^5}(187.9+460)} = \frac{72+4193}{0.237+0.338} = 7417\ lbm$$

$$P_{ww} = \frac{M_{a,ww}RT_{ww}}{V_{ww}} + P_{sat}(T_{ww}) = \frac{7417 \cdot 53.34 \cdot (187.9+460)}{144 \cdot 1.02\times10^5} + 8.9 = 26.4\ psia$$

This hand calculated containment pressure of 26.4 psia (11.7 psig) agrees well with the FSAR value of approximately 12 psig indicated by Figure 6.2-27 and the value of 12.4 psig calculated with the SHEX benchmark analysis.

Extended Power Uprate Effect

A comparison of SHEX analyses performed at the FSAR power level to results obtained at the extended power uprate level are also provided to show the effect due to the power increase. Inputs to the SHEX analyses used for this comparison are based on the current plant geometry taken from the extended power uprate analyses, and include minor geometry input updates (to make the calculation specific to Unit 2). Also, the Technical Specifications value for the RHR flow rate of 7700 gpm and the 1979 ANS 5.1 decay heat with a 2σ adder were used. Additionally, the inputs are designed to minimize the containment pressure as described in the response to NRC Question 59 regarding minimum containment pressure in the NPSH analyses.

The decay heat used in the analysis was based on the ANSI/ANS-5.1-1979 Decay Heat Standard with a two-sigma statistical uncertainty adder. The standard adjustment for neutron capture effects (G-factor) was used. Core design inputs (exposure, residence time and enrichment) were chosen to be representative of current and future fuel cycles. The decay heat table determined in this manner is definitely an upper-bound (i.e., it would not be exceeded) for a variety of different fuel cycles, as long as the exposure and residence time are not greater than the chosen inputs and the enrichment is not substantially less than its chosen value. The time points and interpolation method (linear) were chosen to yield an accurate table, as well as one which is an upper-bound to all tables with more points.

In the generation of the decay heat table, consideration was given to NRC Information Notice 96-039, which points out the variations in decay heat that may be obtained from the same standard, depending upon the choice of input parameters (especially, R, the actinide parameter, and φ, the neutron capture effect parameter). In this application of the standard, the values of R and φ were taken from detailed lattice evaluations based on the exposure and enrichment inputs.

A summary of the analyses results with FSAR power and extended power uprate, including the peak suppression pool temperature and the containment pressure, is shown in Table 56-2

CONCLUSIONS

SHEX gives comparable results to those reported in the FSAR when the same input assumptions are used. Differences in the pressure calculation during the first 10,000 sec of the event analysis are due to the simplifying assumption made in original FSAR analysis that the drywell temperature is instantaneously equal to the spray temperature at the onset of the sprays. This benchmark comparison shows that SHEX is valid code for use in the Plant Hatch extended power uprate calculations.

REFERENCES

1. NEDO-21052, Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels," GE, September 1975.

2. General Electric (GE) Document ID 761E292BA Rev. 8, "Residual Heat Removal System," Process Diagram for Plant Hatch Unit 2, May, 1992.

TABLE 56-1

SUMMARY OF KEY RESULTS FROM FSAR AND SHEX CONTAINMENT RESPONSES

FSAR Comparison	FSAR	SHEX
Peak Pool Temp [°F]	209.8 (1)	210.3
Time of Peak Pool Temp [sec]	≈32,000 (2)	32,000
WW Press @ Time of Peak Pool Temp (3) [psig]	≈12 (2)	12.4

1. This value is the bounding peak suppression pool temperature reported in the FSAR for all four FSAR cases. The case used as the benchmark case should be the limiting configuration with respect to maximum suppression pool temperatures.

2. Time of peak pool temperature taken from Figure 56-1; wetwell pressure at time of peak pool temperature taken from Figure 56-2.

3. The wetwell pressure that exists at the time of the peak suppression pool temperature is of importance because the NPSH margin is at a minimum when the maximum suppression pool temperature occurs. The secondary wetwell pressure peak also occurs near this time and is not significantly different in value.

TABLE 56-2

EFFECT OF POWER UPRATE ON CONTAINMENT RESPONSE

Power Uprate Comparison	**FSAR Power**	**Extended Power Uprate**
Peak Pool Temp [°F]	198.2	206.1
Time of Peak Pool Temp [sec]	24821	27465
WW Press @ Time of Peak Pool Temp [(1)] [psig]	7.7	9.5

1. The wetwell pressure that exists at the time of the peak suppression pool temperature is of importance because the NPSH margin is at a minimum when the maximum suppression pool temperature occurs. The secondary wetwell pressure peak also occurs near this time and is not significantly different in value.

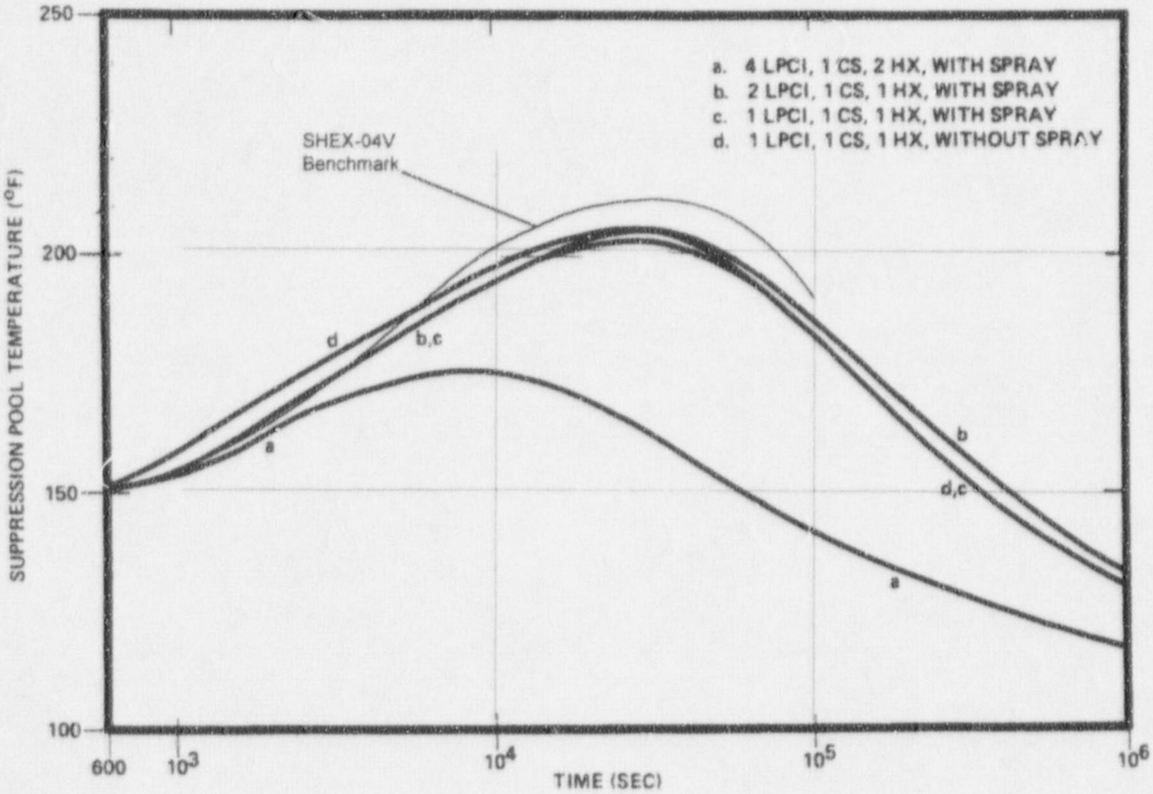


FIGURE 56-1

COMPARISON OF SUPPRESSION POOL TEMPERATURE RESPONSE (UNIT 2 FSAR FIGURE 6.2-29 OVERLAID WITH SHEX BENCHMARK OF CASE C)

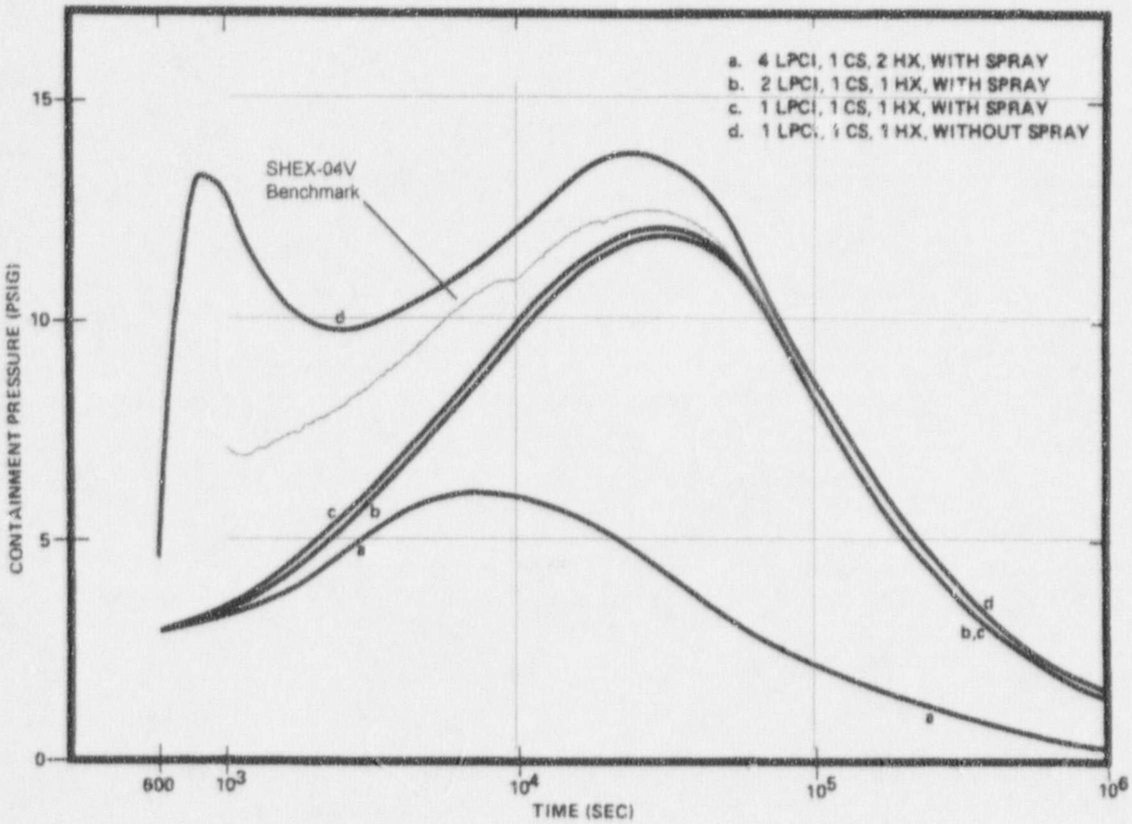


FIGURE 56-2

COMPARISON OF CONTAINMENT PRESSURES (UNIT 2 FSAR FIGURE 6.2-27 OVERLAID WITH SHEX BENCHMARK OF CASE C)

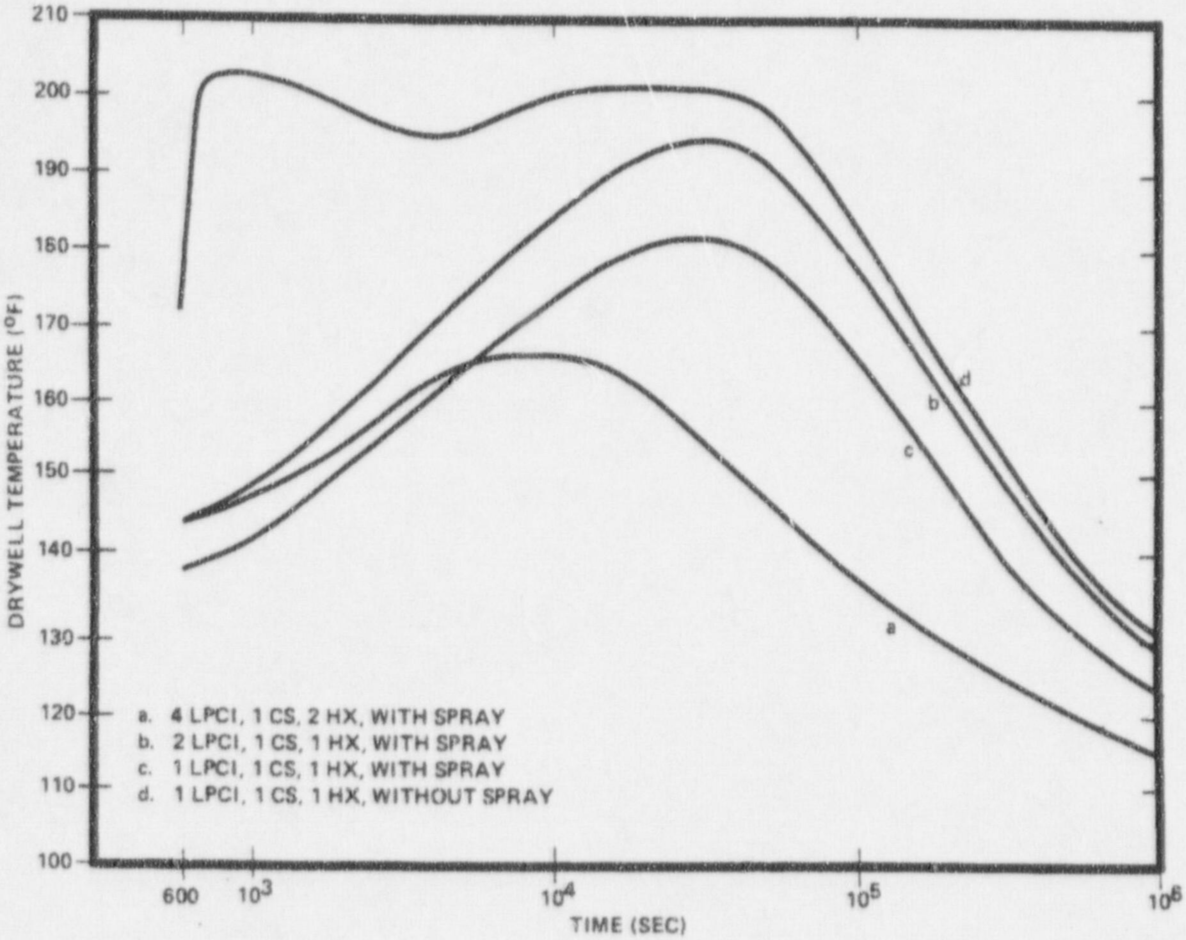


FIGURE 56-3

DRYWELL TEMPERATURE RESPONSES
(UNIT 2 FSAR FIGURE 6.2-28)

NRC QUESTION 59

For review of available containment pressure for the net positive suction head (NPSH), please provide the key assumptions used for the minimum containment pressure analyses. Also provide the updated containment analyses pressure and temperature curves.

SNC RESPONSE

The limiting available net positive suction head (NPSH) for the residual heat removal (RHR) pumps and core spray (CS) pumps corresponds to the minimum suppression chamber pressure with the maximum suppression pool temperature for the most limiting short and long-term loss-of-coolant accident (LOCA) events. The LOCA short-term response occurs during the first 10 min of the LOCA event when no credit is taken for operator actions to control pump flows or initiate containment cooling. The LOCA long-term response includes the time period after 10 min and past the time of the peak suppression pool temperature when it is assumed that the operator controls pump flows and initiates containment cooling.

Model Description

The GE computer code SHEX is used to perform the analysis of the containment pressure and temperature response. The SHEX code has been validated in conformance with the requirements of the GE Engineering Operating Procedures (EOPs).

SHEX uses a coupled reactor pressure vessel and containment model, based on the Reference 1 and Reference 2 models which have been reviewed by the NRC, to calculate the transient response of the containment during the LOCA. This model performs fluid mass and energy balances for the reactor primary system and the suppression pool, and calculates the reactor vessel water level, the reactor vessel pressure, the pressure and temperature in the drywell and suppression chamber airspace, and the bulk average suppression pool temperature. The various modes of operation of all important auxiliary systems, such as safety relief valves (SRVs), the main steam isolation valves (MSIVs), the emergency core cooling systems (ECCS), the RHR, and feedwater, are modeled. The model can simulate actions based on system setpoints, automatic actions and operator-initiated actions.

Short-Term Analysis

The suppression pool temperature and suppression chamber airspace pressure responses to the DBA LOCA were analyzed for a postulated break in the recirculation discharge line with all 4 LPCI pumps and 2 CS pumps available for injection. It is assumed for this analysis that two LPCI pumps inject flow into the broken recirculation loop and subsequently directly into the drywell. This event results in minimum suppression chamber airspace pressures and maximum suppression pool temperatures during the first 10 min of an accident when operator actions are not credited. This event is therefore considered to be limiting with respect to NPSH margins for the first 10 min of the accident.

Although a recirculation discharge line break was modeled for this analysis, the results will be the same for a recirculation suction line break. This is true because for either break location, the break size is sufficiently large such that the break flows for this event are established by the ECCS pump injection flow rate. The discharge break is large enough that the vessel is fully depressurized before the ECCS pumps begin injecting. Because the CS pump flow into the reactor pressure vessel (RPV) and the LPCI pump flow into the broken loop are the same with either break location, the break flows into the drywell will be the same. Consequently, the drywell and suppression chamber airspace pressure and temperature response will be the same for both the recirculation and discharge break locations.

Short term analysis assumptions:

1. With a signal for LPCI initiation, all four RHR pumps start in the vessel-injection mode. Two LPCI pumps inject directly into the drywell. To minimize the drywell pressure response, a conservatively large flow rate with respect to the RHR pump runout flows was assumed.

2. After receiving a signal for CS initiation, the 2 CS pumps are injecting into the vessel for the first 10 min of this event. The runout flow for the CS pump is assumed.

Long-Term Analysis

The long-term analysis assumes a double-ended recirculation suction line break with no off-site power and the assumed failure of one diesel generator. For this case, there is one division of the RHR system available for long-term containment cooling consisting of one heat exchanger, one pump, and two RHR service water (RHRSW) pumps. One loop of the CS system is also available. This containment cooling configuration is the limiting configuration with respect to maximum suppression pool temperature. Therefore, this event is considered to be limiting with respect to NPSH margins for the long-term analysis.

Long-Term Analysis Assumptions

1. With a signal for LPCI initiation, one RHR pump automatically starts in the vessel-injection mode for the first 10 min of the event, when operator actions are not credited. The rated RHR pump flow is assumed.

2. After receiving a signal for CS initiation, the one CS pump is injecting into the vessel. The runout flow for the CS pump is assumed for the entire event.

3. At 10 min into the event, it is assumed that the operator initiates containment cooling. The RHR pump flow is re-routed through one RHR heat exchanger with two RHRSW pumps to the drywell and wetwell sprays. The rated RHR pump flow rate is assumed.

ANALYSIS INPUTS AND ASSUMPTIONS

Input assumptions are used which maintain the overall conservatism in the evaluation by maximizing the suppression pool temperature and conservatively minimizing the suppression chamber airspace pressure and, therefore, minimize the available NPSH. The key input assumptions which are used in performing the Plant Hatch containment LOCA pressure and temperature response analysis are described below.

1. The reactor is assumed to be operating at 102% of the proposed extended power uprate operating power level of 2763 MWt.

2. Vessel blowdown flow rates are based on the Homogeneous Equilibrium Model (Reference 3).

3. The core decay heat is based on ANSI/ANS-5.1-1979 decay heat with 2σ uncertainty adders (Reference 4). Inputs for enrichment, exposure, and residence time that bound the Plant Hatch extended power uprate core design are used. The standard G-factor is used in the calculation of the decay heat curve.

4. Feedwater flow into the RPV continues until all hot feedwater, which maximizes the suppression pool temperature, is injected into the vessel.

5. Thermodynamic equilibrium exists between the liquids and gases in the drywell. Mechanistic heat and mass transfer between the suppression pool and the suppression chamber airspace are modeled to minimize the suppression chamber airspace pressure and temperature.

6. Heat transfer from break fluids to the drywell atmosphere is adjusted to minimize the suppression chamber airspace pressure. For the short-term analysis, 100% of the non-flashing liquid break flow is assumed to be held up in the drywell and to be fully mixed with the drywell fluids before flowing to the suppression pool. The break flow in the short-term case is the LPCI flow into the broken loop, which has a lower temperature than the drywell air. For the long-term analysis, 20% of the non-flashing liquid break flow is assumed to be fully mixed with the drywell fluids. The break flow in the long-term case comes from the vessel and has a higher temperature than the drywell air. Thermal equilibrium conditions are imposed between this held-up liquid and the fluids in the drywell as described in Assumption 5. The liquid not held up is assumed to flow directly to the suppression pool without heat transfer to the drywell fluids.

7. The vent system flow to the suppression pool consists of a homogeneous mixture of the fluid in the drywell.

8. The initial suppression pool volume is at the minimum Technical Specifications (TS) limit to maximize the calculated suppression pool temperature. The minimum initial suppression pool volume is unit specific.

9. The initial drywell and suppression chamber airspace pressures are 16.45 psia.

10. An initial bulk average drywell temperature of 135°F and a relative humidity of 100% are used to minimize the initial non-condensable gas mass and minimize the long-term containment pressure for the NPSH evaluation. The drywell airspace volume is unit specific.

11. The initial suppression pool temperature is at the maximum TS value (100°F) to maximize the calculated suppression pool temperature.

12. The initial suppression chamber airspace temperature is at 100°F and the initial relative humidity is at 100%.

13. The RHR service water temperature is at the maximum allowable value of 95°F to maximize the calculated suppression pool temperature.

14. Heat sinks are used for the short-term analysis to minimize the chamber airspace pressure. Heat sink inputs for these cases were developed based on the Plant Hatch Unit 1 drywell and torus geometry parameters which were compiled and used during the Mark I Containment Long-Term Program. The drywell and torus airspace shell film coefficient is based on the Uchida correlation with a 1.2 multiplier. Condensation heat transfer is assumed at all times, unless the structural

temperature of the heat sink is greater than the airspace saturation temperature in which case natural convection heat transfer is assumed.

15. Heat sinks are not used for the long-term analysis to maximize the suppression pool temperature. This is justified since in the long-term, with drywell and suppression chamber sprays operating, heat sinks have negligible effect on suppression chamber airspace pressure.

16. All CS and RHR Cooling system pumps have 100% of their horsepower rating converted to a pump heat input which is added either to the RPV liquid or suppression pool water. The CS pump heat is unit specific.

17. Heat transfer from the primary containment to the reactor building is conservatively neglected.

18. Containment leakage is included in the calculation of containment pressure response. Including containment leakage has no impact on the peak suppression pool temperature, but will slightly reduce the calculated containment pressure. The leakage assumed was the Plant Hatch Technical Specifications limit of 1.2 % per day.

19. Drywell and suppression chamber sprays with 100% thermal mixing efficiency between the spray liquid and the drywell and suppression chamber atmosphere were assumed.

Based on the above discussions, it is concluded that the containment analyses performed for Plant Hatch Units 1 and 2 with the SHEX computer code have used initial conditions and analysis assumptions appropriate to conservatively minimize containment pressure and maximize suppression pool temperature for use in NPSH evaluations.

Figures 59-1, 59-2 and 59-3 provide the updated containment analyses pressure and temperature curves.

REFERENCES

1. NEMD-10320, "The GE Pressure Suppression Containment System Analytical Model," GE, March 1971.

2. NEDO-20533, "The General Electric Mark III Pressure Suppression Containment Analytical Model," GE, June 1974.

3. NEDO-21052, Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels," GE, September 1975.

4. "Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, Approved by the American National Standard Institute, August 29, 1979.

FIGURE 59-1

COMPARISON OF SUPPRESSION POOL TEMPERATURES

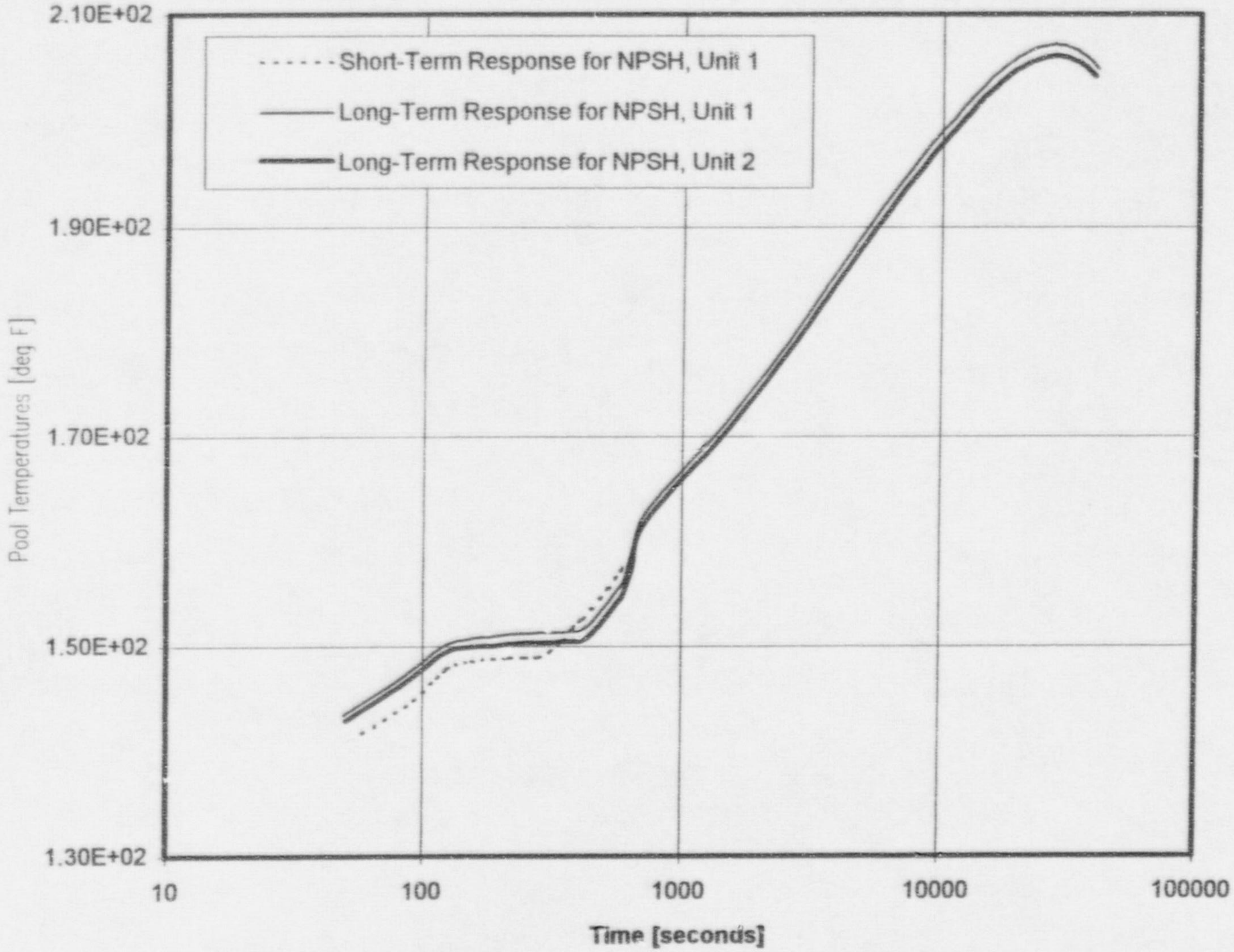


FIGURE 59-2

COMPARISON OF WETWELL TEMPERATURES

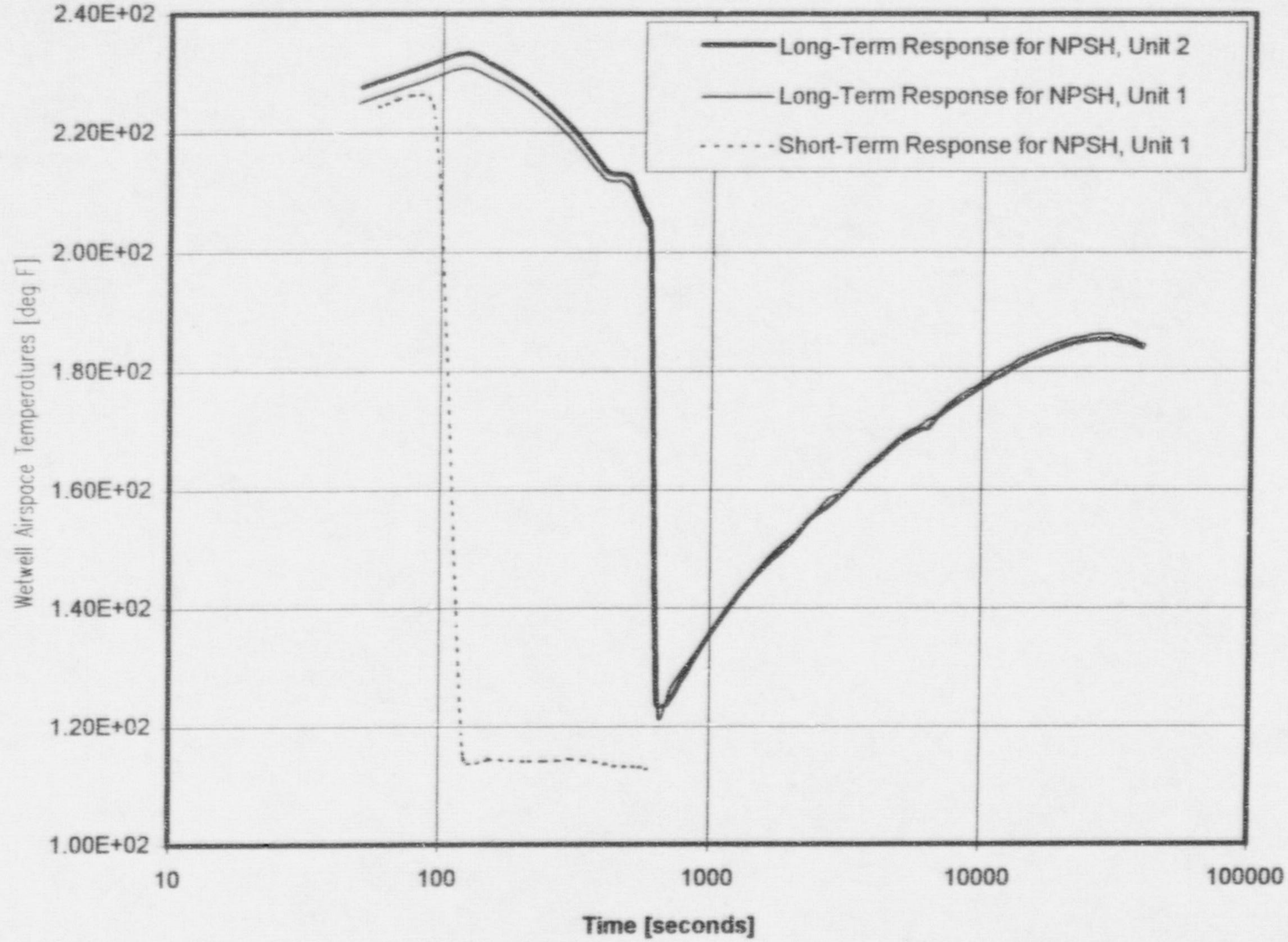
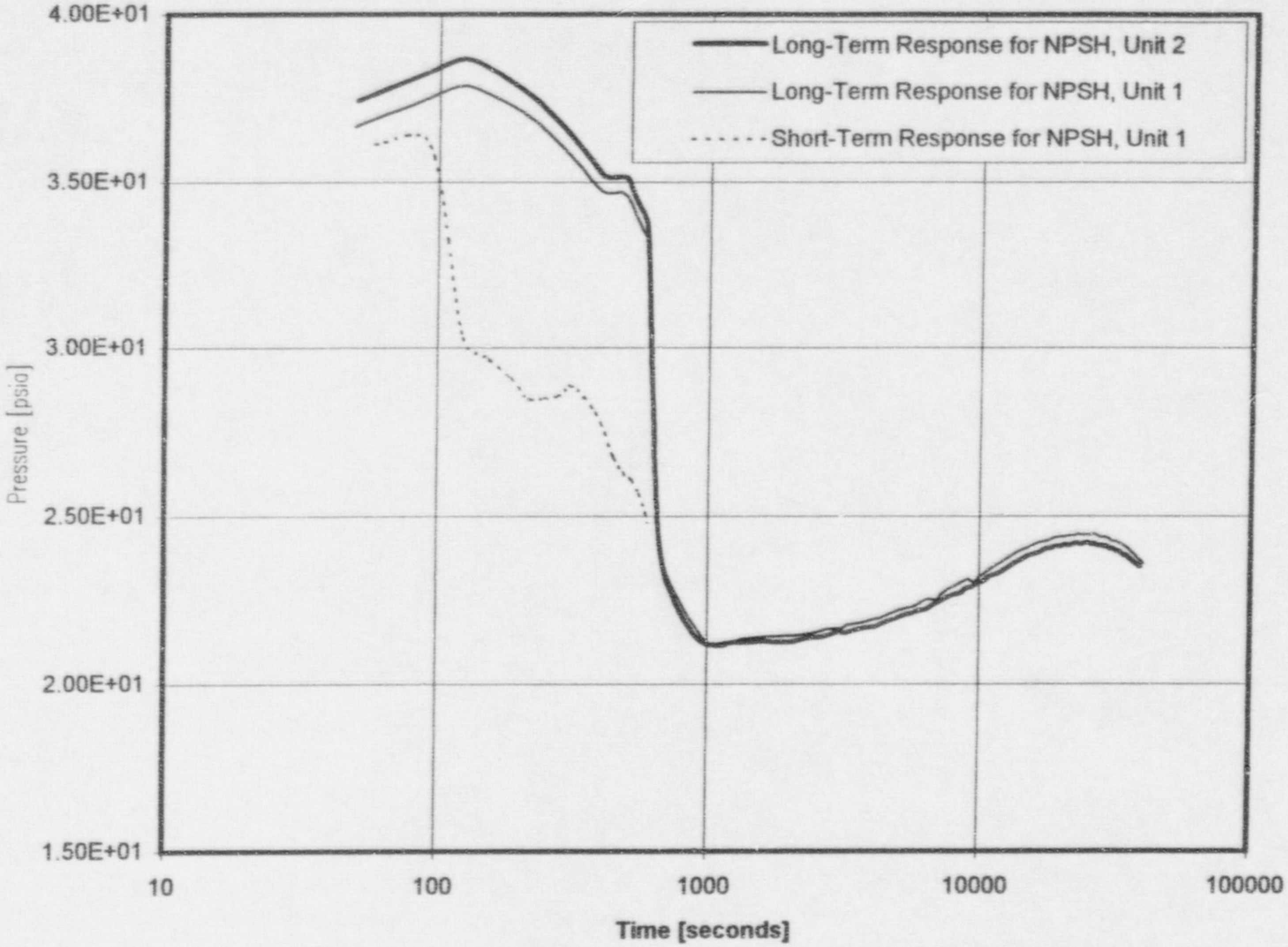


FIGURE 59-3

COMPARISON OF WETWELL PRESSURES



NRC QUESTION 60

For review of used overpressure for Unit 1, please provide the NPSH calculations for residual heat removal and containment [core] spray pumps. The results are tabulated on Page E-5 of your 90-day response to Generic Letter 97-04, dated December 39, 1997.

SNC RESPONSE

SNC's response to Generic Letter 97-04 provided the ECCS pumps NPSH requirements for the proposed extended power uprate conditions. It was SNC's intent to incorporate the most limiting requirements of current licensing activities that would impact the NPSH requirements and thereby provide a bounding response for the current license power level. The following provides a discussion of the NPSH calculations used to support the Generic Letter 97-04 response, as well as the NPSH requirements for both current and extended power uprate conditions. The discussion of the extended power uprate condition also incorporates the impact of the new ECCS suction strainers.

Generic Letter 97-04 Response

As indicated in the referenced 90-day response to Generic Letter 97-04, the NPSH available for each of the RHR and CS pumps for Unit 1 under long-term conditions was determined using a peak suppression pool temperature of 208°F (calculated by GE at extended power uprate conditions of 2763 MWt) and assuming a containment overpressure of 5 psig in order to maintain an approximate 5 psi margin between the overpressure taken credit for and the minimum overpressure of 9.7 psig (also calculated by GE at extended power uprate conditions of 2763 MWt) in accordance with Unit 1 FSAR Section 14.4.3.3. The attached NPSH margin calculations for the Unit 1 RHR and CS pumps (Enclosure 5) utilize these same assumptions for peak pool temperature and overpressure credit. The enclosed calculations represent a bounding case whereby the worst case suction piping configuration and NPSH required were used to determine the worst case NPSH available for each set of ECCS system pumps in lieu of a separate calculation for each pump. The information supplied in the Generic Letter response represents the results of preliminary calculations using the identical methodology to provide an indication of the NPSH available for each pump.

NPSH Requirements (Current Power Level)

The calculation of NPSH available for the Unit 1 RHR and CS pumps at the current power uprate conditions of 2558 MWt considers a peak suppression pool temperature of 202°F and a minimum overpressure of approximately 8 psig. Table 60-1 reflects a bounding case whereby the worst case suction piping configuration and NPSH required for rated flow were used to determine the worst case NPSH available for each set of ECCS system pumps. This information reveals that only a small portion of the containment overpressure available at the time of peak pool temperature is actually required to ensure adequate NPSH for the Unit 1 ECCS pumps. No overpressure is required for Unit 2.

As shown in Table 60-1, the containment overpressure required to assure adequate NPSH for the RHR and CS pumps is 0.88 psi and 0.75 psi respectively. The historical discussion provided in the Unit 1 FSAR, Section 14.4.3.3.1 on the minimum containment pressure required to provide adequate NPSH for the RHR pumps remains valid. This section states: "There is a period when a containment pressure of less than 1 psig is required to provide adequate NPSH for the LPCI pumps."

The Unit 1 FSAR section also states: "The CS pumps have adequate NPSH even if the containment is at atmospheric pressure." Because of the impact of more conservative debris generation and loading requirements for the ECCS suction strainers, the 1 psig containment pressure credit for the RHR pumps is also required for the CS pump at the current power level.

NPSH Requirement (Extended Power Uprate)

Also, in response to Question No. 59 and various discussions with the NRC staff, updated pressure and temperature analyses were performed for extended power uprate with revised assumptions. As the updated pressure and temperature response curves indicate, the peak suppression pool temperature for Unit 1 is approximately 207°F instead of the 208°F resulting from previous analyses. The NPSH available for the RHR and CS pumps at the revised peak pool temperature is shown in the Table 60-2.

As indicated in Table 60-2, a small portion of the available containment pressure is required to ensure adequate NPSH for the Unit 1 RHR (2.1 psi) and CS (2.0 psi) pumps. SNC is requesting an additional five feet (5 ft) of margin, over and above that required, to ensure adequate NPSH for the ECCS pumps to account for potential future issues.

Figure 60-1 and 60-2 show the NPSH margin profiles for both the RHR and CS pumps respectively. Containment overpressure is needed to assure available NPSH from approximately 3 hours until nearly 17 hours into the event. The requested additional margin of five feet would start approximately 1.5 hours into the event and would no longer be necessary after 26.5 hours. From examination of Figures 60-1 and 60-2, containment overpressure plus the requested additional margin is available throughout the event. SNC is therefore requesting containment overpressure credit of 10 feet throughout the event for simplification of Unit 1 NPSH available considerations.

TABLE 60-1

UNIT 1 ECCS PUMPS NPSH MARGIN CURRENT POWER LEVEL

ECCS System	NPSH Required (ft)	NPSH Available (ft)	NPSH Margin (ft)	Overpressure Available (ft)	Overpressure Available (psi)	Overpressure Required (ft)	Overpressure Required (psi)
RHR	15	12.92	-2.08	19.42	8.1	2.08	0.88
CS	14	12.21	-1.79	19.42	8.1	1.79	0.75

TABLE 60-2

CALCULATION OF NPSH AVAILABLE FOR THE UNIT 1 RHR AND CS PUMPS AT THE REVISED PEAK POOL TEMPERATURE

ECCS System	NPSH Required (ft)	NPSH Available (ft)	NPSH Margin (ft)	Overpressure Available (ft)	Overpressure Required (ft)	Overpressure Required (psi)	Overpressure Requested (ft)	Overpressure Requested (psi)
RHR	15	10	-5	22.6	5	2.1	10	4.2
CS	14	9.2	-4.8	22.6	4.8	2.0	10	4.08

FIGURE 60-1

UNIT 1 RHR PUMP NPSH EVALUATION

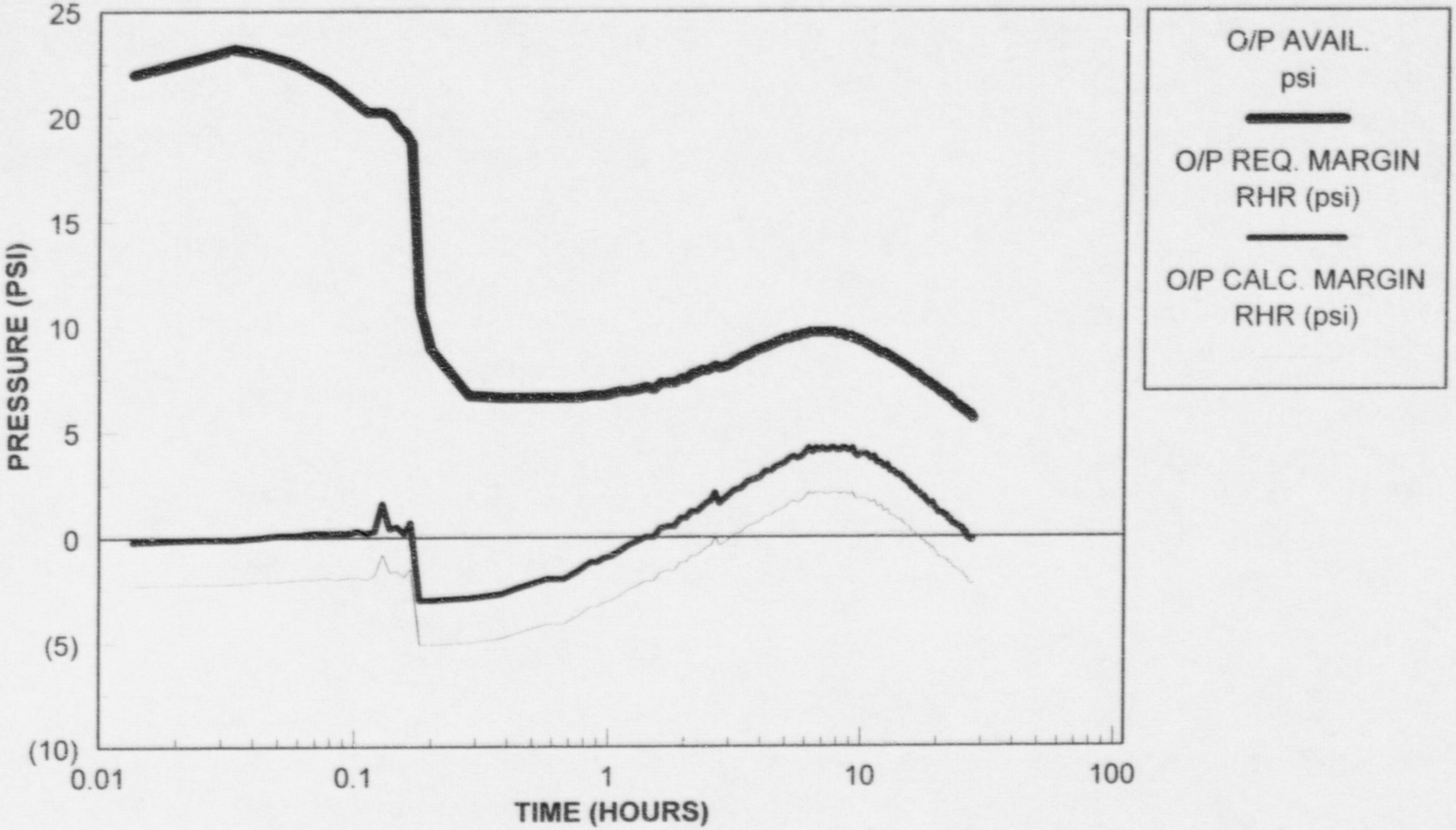
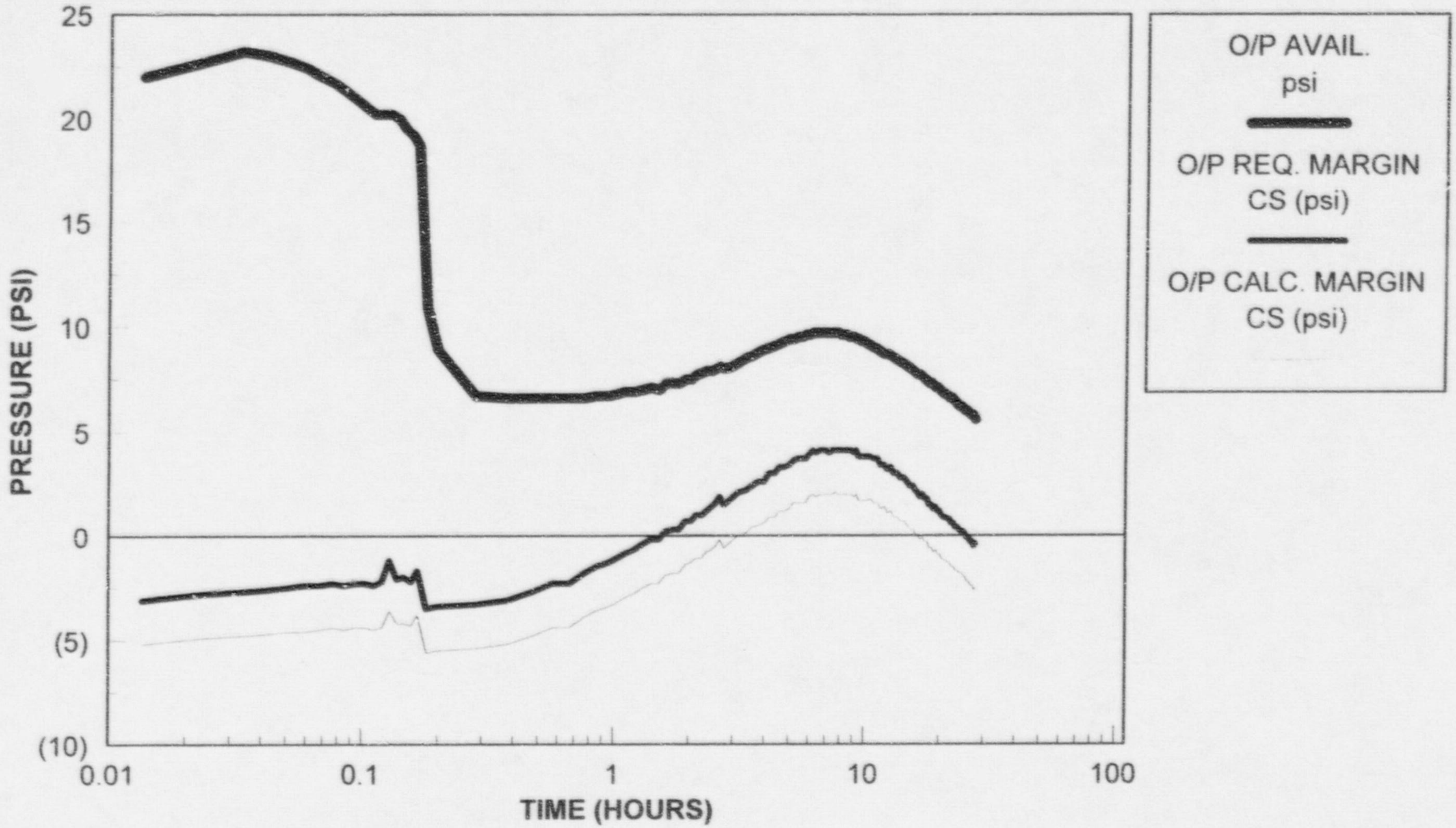


FIGURE 60-2

UNIT 1 CS PUMP NPSH EVALUATION



ENCLOSURE 3

GENERAL ELECTRIC COMPANY

AFFIDAVIT

·. · ··

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the attachment to GE letter GEH082, C. H. Stoll to Tim Long (SNOC), *Transmittal of GE Proprietary SHEX Model Description*, dated June 26, 1998. Attachment A *SHEX Model Description* (GE Proprietary Information) is the proprietary information document. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by bars in the margin is classified as proprietary because it contains detailed design bases and methods and processes regarding the use of analytical models, including computer codes, which GE has developed or modified, and applied to perform evaluations of containment pressurization and heat transfer capability for loss-of-coolant accidents for the BWR. This detailed level of information normally only is available for GE internal use, is not supplied even to our customers, and only is available for audit by customers and the NRC. This information shows in specific detail the processes, codes and methods employed to perform the evaluations.

The development and modification of this information and models for these BWR analysis computer codes was achieved at a significant cost, on the order of several hundred thousand dollars, to GE.

The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed or modified, and applied to perform evaluations of containment pressurization and heat transfer capability for loss-of-coolant accidents for the BWR. The development and modification of this overall approach was achieved at a significant additional cost to GE, over and above the cost of developing the underlying individual proprietary analyses.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas <u>not</u> so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 26th day of June 1998.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 26 day of JUNE 1998.

Lesley E. Gourvitch

Notary Public, State of California

LESLEY E. GOURVITCH
Comm. # 1046603
NOTARY PUBLIC - CALIFORNIA
Santa Clara County
My Comm. Expires Dec. 23, 1998

State of California
County of Santa Clara

On 6/26/98 before me, Lesley E. Gourvitch Notary Public
(DATE) (NAME/TITLE OF OFFICER-i.e."JANE DOE, NOTARY PUBLIC")

personally appeared George B. Strambach
(NAME(S) OF SIGNER(S))

☐ personally known to me -OR- ☒ proved to me on the basis of satisfactory evidence to be the person(s) whose name(s) is/are subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their authorized capacity(ies), and that by his/her/their signature(s) on the instrument the person(s), or the entity upon behalf of which the person(s) acted, executed the instrument.

PWR1 LESLEY E. GOURVITCH
Comm. # 1046603
NOTARY PUBLIC - CALIFORNIA
Santa Clara County
My Comm. Expires Dec. 23, 1998 PWR1

Witness my hand and official seal.

(SEAL)

Lesley E Gourvitch
(SIGNATURE OF NOTARY)

RIGHT THUMBPRINT (Optional)

TOP OF THUMB HERE

CAPACITY CLAIMED BY SIGNER(S)
☐INDIVIDUAL(S)
☐CORPORATE ____

OFFICER(S) ____
(TITLES)

☐PARTNER(S) ☐LIMITED
☐GENERAL
☐ATTORNEY IN FACT
☐TRUSTEE(S)
☐GUARDIAN/CONSERVATOR
☐OTHER: ____

SIGNER IS REPRESENTING:
(Name of Person(s) or Entity(ies)

RIGHT THUMBPRINT (Optional)

TOP OF THUMB HERE

CAPACITY CLAIMED BY SIGNER(S)
☐INDIVIDUAL(S)
☐CORPORATE ____

OFFICER(S) ____
(TITLES)

☐PARTNER(S) ☐LIMITED
☐GENERAL
☐ATTORNEY IN FACT
☐TRUSTEE(S)
☐GUARDIAN/CONSERVATOR
☐OTHER: ____

SIGNER IS REPRESENTING:
(Name of Person(s) or Entity(ies)

ATTENTION NOTARY

The information requested below and in the column to the right is **OPTIONAL**. Recording of this document is not required by law and is also optional. It could, however, prevent fraudulent attachment of this certificate to any unauthorized document.

THIS CERTIFICATE MUST BE ATTACHED TO THE DOCUMENT DESCRIBED AT RIGHT:

Title or Type of Document: Affidavit

Number of Pages: 5 Date of Document: 6/26/98

Signer(s) Other Than Named Above ____

7 67775 63237 8

ENCLOSURE 4

SNC RESPONSE TO ADDITIONAL NRC QUESTIONS

Enclosure 4

Edwin I. Hatch Nuclear Plant
Request for Additional Information:
Extended Power Uprate License Amendment Request

SNC Response to Additional NRC Questions

NRC QUESTION 86

Did the Hatch GL 89-10 MOV program reviewed by the NRC inspectors in 1995 include the power uprate conditions associated with the proposed Tech Spec amendment?

SNC RESPONSE

No, the GL 89-10 program was reviewed by the NRC before implementation of the initial power uprate. The motor-operated valve (MOV) program has since been updated to reflect the impact of the initial power uprate. Since the initial power uprate was accomplished by an increase in reactor pressure, the differential pressure calculations for the affected valves were revised. Revision 4 of the Torque Switch Setting Guide incorporated the results of the differential pressure calculations as a result of the initial power uprate.

At the request of the NRC Staff reviewer, Table 86-1 provides to give a summary of the MOV program valves identified as being affected during the initial power uprate program evaluation. Table 86-1 therefore represents a list of MOV changes due to the initial power uprate that were not part of the NRC inspection in 1995 of the Plant Hatch MOV program. The MOV program is a controlled program with periodic changes due to other issues and efforts, therefore Table 86-1 does not and is not intended to represent the current status of the MOV program.

As a result of the extended power uprate program, there will not be an increase in any system operating pressure, reactor pressure or safety relief valve (SRV) setpoints. The differential pressures used to determine valve/operator thrust and torque requirements are not affected by extended uprate conditions. Therefore no MOV's are affected and the MOV program is not required to be updated as a result of the extended power uprate program.

TABLE 86-1

GENERIC LETTER 89 10
MOTOR-OPERATED VALVES AFFECTED BY POWER UPRATE
AFTER SCREENING

Unit	MPL#	Size	Active Safety Function	Valve Description	Valve Type	Increase in Thrust (%)	Increase in Torque (%)
1/2	B21-F016	3"	C	Main steam line drain isolation	Gate	2	3
1/2	B21-F019	3"	C	Main steam line drain isolation	Gate	2	2
1/2	E41-F001	10"	O	HPCI turbine steam supply	Gate	3	3
1/2	E41-F002	10"	C	HPCI steam supply inboard isolation	Gate	3	3
1/2	E41-F003	10"	C	HPCI steam supply outboard isolation	Gate	3	3
1/2	E51-F045	4"	O	RCIC turbine steam supply	Globe	2	3
1/2	E51-F007	4"	C	RCIC steam supply inboard isolation	Gate	2	3
1/2	E51-F008	4"	C	RCIC steam supply outboard isolation	Gate	2	3
1/2	G31-F001	6"	C	RWCU inboard isolation	Gate	3	3
1/2	G31-F004	6"	C	RWCU outboard isolation	Gate	3	3
1/2	E41-F006	14"	O	HPCI pump discharge	Gate	3	3
1/2	E51-F013	4"	O	RCIC pump discharge	Gate	3	3

NRC QUESTION 87

What is the meaning of 'insignificant increase' for the pressure and temperature of the RRS system in Table 30-1 of the licensee's March 9, 1998 response to the staff's RAI?

SNC RESPONSE

Increasing the reactor core thermal power from 2558 MWt to 2763 MWt will result in main steam and feedwater flow increases. The increased steam generation in the reactor core, at the same core mass flow rate, will produce a small increase in the core and steam separator pressure drops. Calculations show that the increased steam generation in the core increases the average core exit quality by about 1.1%. This results in an increase in the flow losses of the internal flow loop of about 0.5 psi. If no adjustment is made to the recirculation pump speed and discharge head, the core flow rate can be expected to decrease by about 1%. Calculations show the recirculation pump discharge head increase required to maintain rated core flow at 2763 MWt will increase by 7 ft of H_2O, or less than 2%, when compared to the value which is currently calculated for 2558 MWt operation.

For the reactor heat balance, General Electric (GE) calculated a small temperature decrease in the recirculation system. This phenomena is due to the increased feedwater flow which is colder compared to the bulk water flow. Thus, the mixture in the downcomer is colder, and so is the recirculation flow. The Plant Hatch Unit 1 calculated recirculation temperature changes from 533°F (2558 MWt) to 532.1°F (2763 MWt). The Plant Hatch Unit 2 calculated recirculation temperature changes from 536°F (2558 MWt) to 534.6°F (2763 MWt). However, the system evaluation ignored this thermodynamic effect to determine an upper bound operating temperature impact on the equipment. Therefore, the table has only stated a slight temperature increase to consider the additional pump heating energy. The calculated enthalpy change due to this heating is less than 0.1 Btu/lb increase which corresponds to a temperature increase of less than 0.1°F.

NRC QUESTION 88

What are the effects on power-operated valve performance (such as from motor ambient temperature or pressure locking/thermal binding) on the changes identified for the Containment Spray (CS) system in Table 30-1?

SNC RESPONSE

Please note, CS in Table 30-1 refers to core spray (CS) not containment spray. As indicated in Table 30-1, there will be a slight increase in the post-loss-of-coolant accident (LOCA) suppression pool temperature and pressure for the CS system due to extended power uprate. The power-operated valves for the CS will be unaffected by the increases. The CS configuration for both Units 1 and 2 are the same. Two lines of the CS are connected to the suppression pool. One is the pump suction line, which contains two power-operated valves. One valve is a normally open fail open air operated butterfly valve. The other is a normally open gate MOV. Neither valve has an active safety function. The other line that connects to the suppression pool is the pump minimum flow line. The minimum flow line has a normally open gate MOV that has an active safety function both to open and to close. The slight increase in suppression pool pressure would act to help the valves by reducing the differential pressure across the valves. The original GL 95-07 (Pressure Locking/Thermal Binding) evaluation for the minimum flow valves was performed at 210°F for Unit 1 and 195°F for Unit 2. The slight increase to 212°F does not change the conclusion found in the original evaluation that the valves are not susceptible to pressure locking or thermal binding, since the evaluation determined the valves to be not susceptible for reasons independent of the suppression pool temperature. A review of the original GL 95-07 evaluation found a total of 36 valves for the two units that were connected to the suppression pool. However, the pressure locking or thermal binding determination for these valves was not dependent on the suppression pool temperature. Table 88-1 provides information on the 36 valves and the evaluation determination.

TABLE 88-1

HATCH UNIT 2 VALVE PRESSURE LOCKING OR THERMAL BINDING EVALUATION DETERMINATION

Valves	Type	Normal Position	Active Safety Function	Pressure Locking	Thermal Binding
1E11F007A	SOLID	OPENED	BOTH	Reason 4	Reason 5
1E11F007B	SOLID	OPENED	BOTH	Reason 4	Reason 5
2E11F007A	FLEX	OPENED	BOTH	Reason 1	Reason 5
2E11F007B	FLEX	OPENED	BOTH	Reason 1	Reason 5
1E11F015A	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E11F015B	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E11F015A	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E11F015B	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E11F021A	SOLID	CLOSED	OPEN	Reason 4	Reason 3
1E11F021B	SOLID	CLOSED	OPEN	Reason 4	Reason 3
2E11F021A	FLEX	CLOSED	OPEN	Reason 2	Reason 3
2E11F021B	FLEX	CLOSED	OPEN	Reason 2	Reason 3
1E11F028A	FLEX	CLOSED	OPEN	Reason 2	Reason 3
1E11F028B	FLEX	CLOSED	OPEN	Reason 2	Reason 3
2E11F028A	FLEX	CLOSED	OPEN	Reason 2	Reason 3
2E11F028B	FLEX	CLOSED	OPEN	Reason 2	Reason 3
1E21F005A	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E21F005B	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E21F005A	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E21F005B	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E21F031A	SOLID	OPENED	BOTH	Reason 4	Reason 5
1E21F031B	FLEX	OPENED	BOTH	Reason 6	Reason 5
1E21F031B	FLEX	OPENED	BOTH	Reason 6	Reason 5
1E21F031B	FLEX	OPENED	BOTH	Reason 6	Reason 5
1E41F006	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E41F006	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E41F041	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E41F041	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E41F042	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E41F042	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E51F013	FLEX	CLOSED	OPEN	Reason 1	Reason 3
2E51F013	FLEX	CLOSED	OPEN	Reason 1	Reason 3
1E51F029	FLEX	CLOSED	OPEN	Reason 2	Reason 3
2E51F029	FLEX	CLOSED	OPEN	Reason 2	Reason 3
1E51F031	FLEX	CLOSED	OPEN	Reason 2	Reason 3
2E51F031	FLEX	CLOSED	OPEN	Reason 2	Reason 3

Reason Code

1. Pressure Locking is not affected by the increase in the suppression pool temperature due to the hole drilled in the disk.

2. Pressure Locking is not affected by the increase in the suppression pool temperature since the valve is not exposed to elevated fluid temperatures until after the valve is opened. Therefore, the valve has performed it active function before any heat-up of the valve.

3. Thermal binding is not affected by the increase in the suppression pool temperature since the valve is not exposed to elevated fluid temperatures until after the valve is opened. Therefore, the valve has performed it active function before any heat-up of the valve.

4. Pressure Locking is not affected by the increase in the suppression pool temperature since the valve is a solid wedge.

5. Thermal binding of the minimum flow valve is not affected by the increase in the suppression pool temperature. The pump will automatically start on a LOCA signal. The injection valve is normally closed. It will open after the vessel pressure drops to 500 psi. The minimum flow valve is normally open. This value does not close until the injection valve opens and injection into the vessel is sufficient to provide a high flow signal. Thus, the minimum flow valve will be heated somewhat by flow from the suppression pool while the valve is in the open position. The valve will automatically close in a heated condition on high flow. The valve will automatically open if the pump were to trip for any reason. The valve will not cool in the closed position before reopening. However, the valve may increase in temperature while closed since the accident will cause the ambient room temperature to increase and cause the suppression pool fluid temperature to continue to increase. Therefore, the valve is not subject to thermal binding.

6. Pressure Locking was assumed not to be a concern since opening of the valve due to pump trip and subsequent restart is unnecessary for the valve to perform its intended function. The function of the valve is to be open when necessary to provide a flow path ensuring that the minimum flow requirements of the pump are met. The valve is normally open, and the system logic does not close the valve until the system flow exceeds the minimum flow requirements. If the pump trips with the minimum flow valve closed, opening of the valve when the pump is restarted is unnecessary. This is because the injection flow path has been established with the injection valve open, and the flow is adequate to meet the minimum flow requirements without the minimum flow path. Therefore, if the minimum flow valve is pressured locked and cannot be reopened, it is of no consequence since there is no longer a requirement

for the valve to be open for minimum flow conditions to be meet. As soon as the pump restarts, a high flow signal would be present, automatically closing the minimum flow valve if it had opened.

NRC QUESTION 89

What is the effect of the pressure drop for the MSIVs that 'exceeds design' as noted in Table 30-1?

SNC RESPONSE

The design requirement of a maximum 7 psi drop across the MSIVs was only an allocation of the pressure drop between the reactor and the turbine inlet conditions. The effect of increased pressure drop on the turbine inlet conditions is expected and accounted for by the turbine generator modifications.

The MSIVs are designed to permit steam flow of 2,740,000 lb/hr at 1005 psig saturated, with the allowable pressure drop of 7 psi. For extended power uprate, the steam flow rate will be 2,886,000 lb/hr for Unit 1 and 2,995,000 lb/hr for Unit 2. Consequently, the pressure drop across the valves will be increased. The valve manufacturer indicated that higher pressure drop across the valve at the extending power uprate condition will not impact valve performance. The only effect of increased pressure drop is decreased inlet pressure to the high pressure turbine which does not affect safety.

NRC QUESTION 90

Did the power uprate involve any changes to environmental temperature or electric power supply that might affect power-operated valve performance?

SNC RESPONSE

No change in the performance of MOVs is anticipated due to environmental temperature changes. There are no changes in the accident temperature profile outside containment; changes inside containment are considered negligible, and the normal service temperatures are expected to increase little, if any.

Also, no change in the performance of MOVs is anticipated due to changes in the electric power supply (including load center, starter, and cable) resulting from extended power uprate. As indicated above, the changes to normal and accident temperature resulting from power uprate will be minimal, if any, and will not exceed the limits established prior to extended power uprate.

NRC QUESTION 91

Section 10.2.1.1 states that the accident temperature profile at extended power uprate conditions exceeds the current accident profile by up to 7°F during the time period from 35,000 sec to 70,000 sec and this will have no effect on qualification of any equipment. For each component on the EQ Master List, does the existing qualification test data envelope the accident temperature profile at extended power uprate conditions with the required margin?

SNC RESPONSE

Yes, the existing qualification test data do envelope the accident temperature profile at extended power uprate conditions.

The qualification test data for each component on the EQ Master List envelopes the extended power uprate accident temperature profile. The drywell temperature profile for extended power uprate shows that the peak temperature under worse case accident conditions is less than 330°F which is the presently assumed peak temperature. Figures 91-1 and 91-2 are the Unit 1 and 2 drywell temperature EQ analysis profiles. GE has re-evaluated the drywell profiles at extended power uprate conditions. The re-analysis results show that the peak drywell temperature under worst-case accident conditions is below 330°F, which is the peak temperature presently assumed in evaluating the adequacy of environmental qualification tests for drywell equipment. Therefore, the extended power uprate accident temperature profile will have no effect on the environmental qualification of equipment.

Degraded equivalency analysis, documented in SCS Calculation SINH 97-004, shows that the present worst case design basis earthquake (DBE) profile envelopes the new accident profile at extended power uprate conditions. Therefore, qualification of the temperature elements to the present worst case DBE profile also demonstrates qualification to the new accident profile under power uprate conditions. For this particular qualification program, it is also evident by observation that the minor change in the accident profile has no impact of qualification. (Margin is required only during the peak of the accident profile. After the peak, margin is not required.)

FIGURE 91-2

HATCH UNIT 2 DRYWELL TEMPERATURE E/Q ANALYSIS

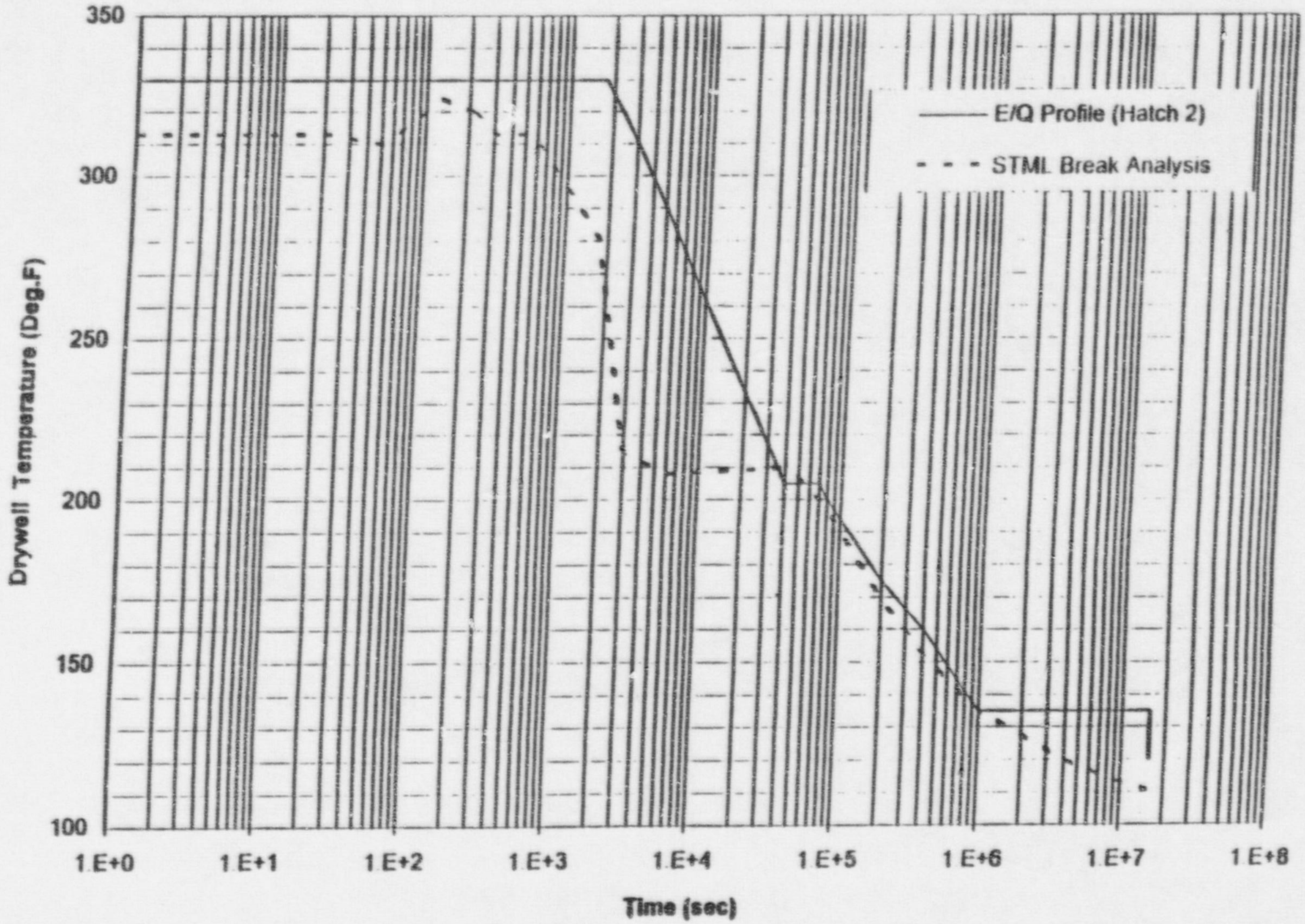
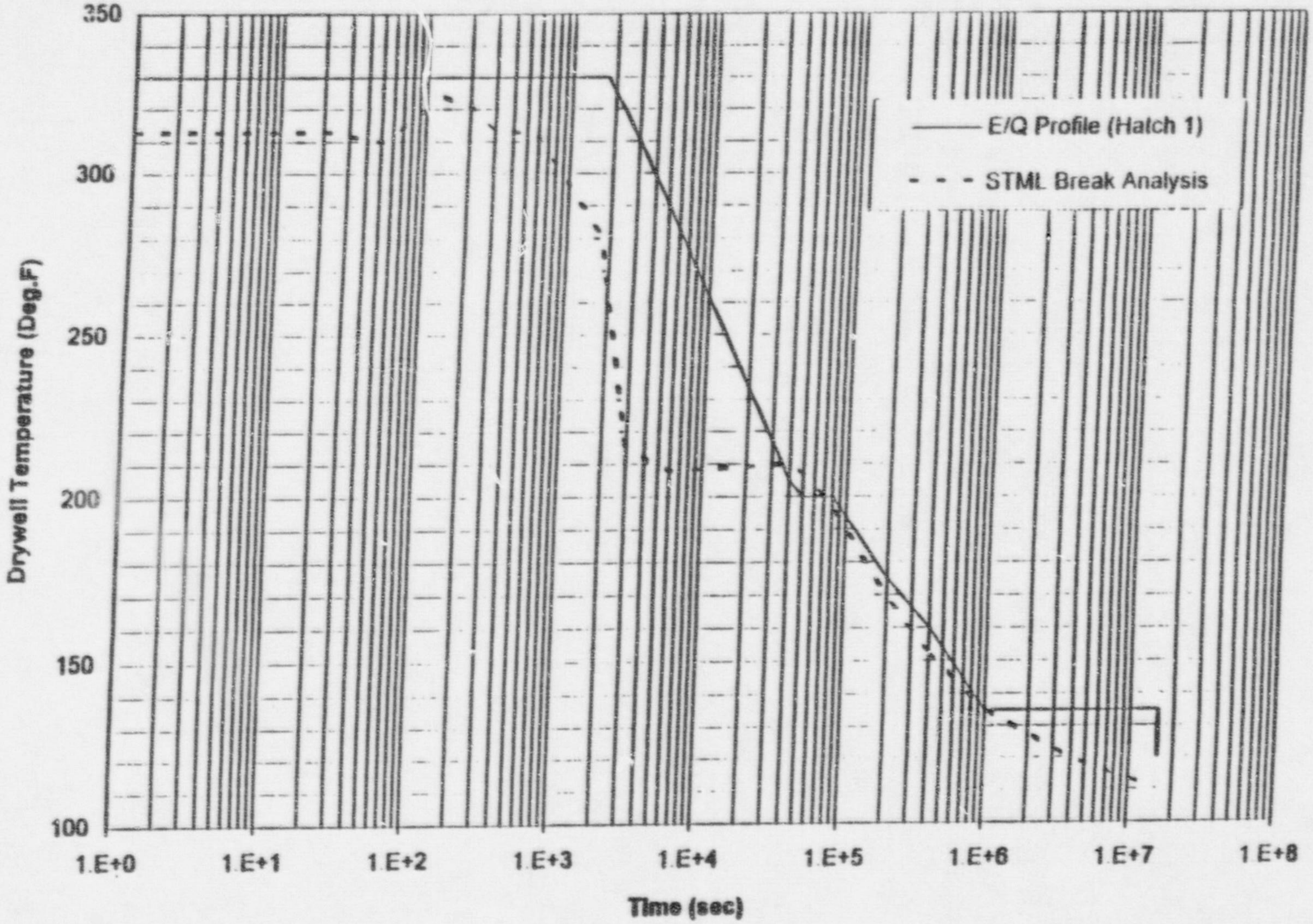


FIGURE 91-1

HATCH UNIT 1 DRYWELL TEMPERATURE E/Q ANALYSIS



NRC QUESTION 92

The P-T curves for Hatch Unit 2 refer to a shift. Does that shift include the delta RTndt and the margin term?

SNC RESPONSE

Yes, the shift includes the delta RTndt and the margin term.

NRC QUESTION 93

In developing the Inservice Hydrostatic and Inservice Leakage Test curves, is the safety factor 1.5 or 2.0 and does the curves actually include a 20°F/hr heat-up/cool-down rate (usually the stress resulting from a 20°F/hr rate is not significant enough to be included)?

SNC RESPONSE

The safety factor used for the Hydrostatic Test curves is 1.5. The 20°F/hr rate is also included, even though the stress is small.

NRC QUESTION 94

In the previous submittal (submittal prior to extended power uprate) weld 10137 was the limiting beltline material for Hatch Unit 2. In the Staff SER of April 4, 1997 for Hatch Unit 2 the 1/4T fluence for the limiting beltline material is reported to be 0.154E18 n/cm^2, however on Table 9-4 of the RAI HL-5579 the 1/4T fluence for 10137 is 0.0947E19. What is the discrepancy?

SNC RESPONSE

The Staff SER of April 4, 1997 approved the license amendment request to revise the Plant Hatch RPV pressure/temperature curves. The analyses performed to support the license amendment assumed a peak f I.D. of 2.17E18 n/cm^2.

For the extended power uprate analyses, the same fluence could have been applied to lower longitudinal weld 10137, Linde 0091/flux lot 3999; however, the conservatism was removed by taking into account the lower axial flux distribution from the neutron transport analysis. The factor was 0.64 times the peak location. Thus; for lower longitudinal weld 10137, Linde 0091/flux lot 3999:

Peak f I.D. = $2.17E18\ n/cm^2 \times .64 = 1.39E18\ n/cm^2$

Peak f I/4T. = $1.39E18\ n/cm^2 \times e^{(-0.24 \times 6.38/4)} = 9.47E17\ n/cm^2$

Also, for the extended power uprate evaluation, the fluence applied to lower-intermediate longitudinal weld 51874, Linde 0091/flux lot 3458 and shown in Table 9-4 of RAI HL-5579 is as follows:

Peak f I.D. = $2.17E18\ n/cm^2$

Peak f I/4T. = $2.17E18\ n/cm^2 \times e(-0.24 \times 5.38/4) = 1.57E18\ n/cm^2$

NRC QUESTION 95

For the USE calculation in Table 10-4 of the RAI HL-5579, page E1-30, the limiting beltline % Cu is 0.23 and the 32 EFPY fluence is $2.17E18\ n/cm^2$. This also seems like a possible inconsistency, is it?

SNC RESPONSE

For the limiting beltline weld USE calculation the practice is to use the limiting % Cu of all the welds (i.e., 0.23 % Cu) together with the highest fluence (i.e., $2.17E18\ n/cm^2$), regardless of which weld was limiting with respect to adjusted reference temperature. Note that although the fluence required for USE calculations is the 1/4T fluence, the surface fluence was conservatively used for the Plant Hatch calculations.

NRC QUESTION 96

Please confirm that the non-beltline curves do not change as a result of extended power uprate.

SNC RESPONSE

The non-beltline curves do not change.

NRC QUESTION 97

Please provide an updated initial RTndt tabulation for the Unit 2 beltline materials, like those of page 6 in the SER.

SNC RESPONSE

The initial RTndt and ART values are tabulated in Table 97-1 below.

TABLE 97-1

HATCH UNIT 2 INITIAL REFERENCE TEMPERATURES/ART VALUES

Lower-Intmed
Thickness = 5.38 inches

32 EFPY Peak I.D. fluence = $2.17E + 18\ n/cm^2$
32 EFPY Peak 1/4 T fluence = $1.57E + 18\ n/cm^2$

Lower
Weld Thickness = 5.38 inches (Girth)
Plate Thickness = 6.38 inches

32 EFPY Peak I.D. fluence = $1.39E + 18\ n/cm^2$
32 EFPY Peak 1/4 T girth weld fluence = $1.01E + 18\ n/cm^2$
32 EFPY Peak 1/4 T plate & long. weld fluence = $9.47E + 17\ n/cm^2$

Component	Weld Type	Heat or Heat/Lot	% CU	% Ni	CF	Initial RTndt °F	32 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
PLATES:												
Lower												
G-6603-1		C8553-2	0.08	0.58	51	-20	20.7	0.0	10.4	20.7	41.4	21.4
G-6603-2		C8553-1	0.08	0.58	51	24	20.7	0.0	10.4	20.7	41.4	65.4
G-6603-3		C8571-1	0.08	0.53	51	0	20.7	0.0	10.4	20.7	41.4	41.4
Lower-Intmed												
G-6602-2		C8554-1	0.08	0.57	51	-20	26.2	0.0	13.1	26.2	52.4	32.4
G-6602-1		C8554-2	0.08	0.58	51	-10	26.2	0.0	13.1	26.2	52.4	42.4
G-6601-4		C8579-2	0.11	0.48	73	-4	37.5	0.0	17.0	34.0	71.5	67.5
WELDS:												
Lower long.	101-842	10137, LINDE 0091	0.23	0.50	154.5	-50	62.7	0.0	28.0	56.0	118.7	68.7
Lower-Intmed long.	101-834	51874, LINDE 0091 / Flux Lot 3458	0.18	0.50	138	-50	70.8	0.0	28.0	56.0	126.8	76.8
Lower to Lower-Int Grith	301-871	4P6052, LINDE 0091 / Flux Lot 0145	0.07	0.03	35.5	-50	14.8	0.0	7.4	14.8	29.7	-20.3

NRC QUESTION 98

Submit the RPV fluence analyses that have been performed for Hatch Units 1 and 2 to support the EOL fluence values cited in Southern Nuclear Company's August 8, 1997 extended power uprate submittal and/or in your March 9, 1998 response to the staff's RAI on the extended power uprate submittal. Your response should include a description of the calculational methodology employed, the use (on non-use) of dosimeter wire analyses in the determination of the best-estimate fluence, and any other pertinent information. Given that RPV surveillance data is being used in the evaluation of Hatch Unit 1, also provide a discussion of how the Unit 1 surveillance capsule fluences were determined.

SNC RESPONSE

The first step in the fluence analysis was to perform a neutron flux analysis to determine the expected increase in fast neutron (Energy > 1 MeV) flux at the reactor pressure vessel due to the increased thermal power associated with extended power uprate operating conditions.

The neutron flux distribution and lead factor were determined for extended power uprate operation by using a combination of two separate two-dimensional neutron transport computer analyses. The lead factor is defined as the ratio of the flux at the capsule to the peak flux at the RPV inside surface. The analyses were performed with the discrete-ordinates transport (DORT) code. The first of these was performed with a model specified in (R,θ) geometry to establish the azimuthal and radial variation of flux at the fuel midplane elevation. The (R,θ) model provides a detailed description of 1/8th of the core geometry in a horizontal slice at midplane. The second analysis was performed in (R,Z) geometry and determined the variation of flux with elevation. The (R,Z) model uses a simplified cylindrical representation of the core and incorporates detailed axial variations of power and coolant density to provide a reasonable approximation to the relative axial flux distribution. The respective two-dimensional distribution results were combined to provide a simulation of the three-dimensional distribution of flux. The analysis for extended power uprate utilized the same model of the physical geometry as used in the previous analysis for the original rated power. Core power distribution, core region compositions, and coolant densities were revised to simulate extended uprate operation. The ratio of fluxes calculated at a given spatial point with the two models provides the predicted change in neutron flux associated with extended power uprate.

The predicted fast neutron flux at the capsule center for the extended uprate was 9.9% higher than the flux calculated for original rated power operation. As a result of the change in shape of the flux distribution, the lead factor changed from 0.62 for original rated power to 0.56 for extended uprate. The best-estimated peak flux was increased an additional 10% to provide additional margin.

Secondly, fluence was calculated for a specific operating period given the increase in neutron flux estimated and the lead factor for extended power uprate conditions. Additional information needed was the fluences from the latest tested RPV surveillance capsule; obtained from Reference 1 for Unit 1 (second capsule at 14.3 EFPY) and Reference 2 for Unit 2 (first capsule at 6.58 EFPY). The EFPY at the start of the 105% and the 113.4% uprates were provided by LaSalle for both Units; the values are 14.3 and 17.0 EFPY, respectively for Unit 1, and 11.25 and 13.9 EFPY, respectively for Unit 2. The equations to calculate end of life peak ID fluences (n/cm^2) are provided below:

Unit 1:

$$f_{17.0} = \{f_{capsule14.3} + 1.1 \times [f_{capsule14.3} \times (17-14.3)/14.3]\}/.62$$

$$f_{32} = f_{17.0} + 1.1 \times 1.099 \times [f_{capsule14.3} \times (32-17)/14.3]/.56$$

Unit 2:

$$f_{11.25} = f_{capsule6.58} \times (11.25/6.58)/.62$$

$$f_{13.9} = f_{11.25} + 1.1 \times [f_{capsule6.58} \times (13.9-11.25)/6.58]/.62$$

$$f_{32} = f_{13.9} + 1.1 \times 1.099 \times [f_{capsule6.58} \times (32-13.9)/6.58]/.56$$

Finally, fluence at 1/4 depth (1/4T) into the vessel wall from the inside diameter was determined using NRC Regulatory Guide 1.99 Revision 2, equation 3 of Paragraph 1.1. This 1/4T depth is recommended in ASME BPV Code Section XI, Appendix G, Subarticle G-2120, as the maximum postulated defect depth.

NRC QUESTION 99

The power upgrade submittal states that the core spray mini-flow valves have an active safety function to open. This conflicts with the October 21, 1996 GL 95-07 submittal which states that the valves are normally open and close on high flow after the core spray injection valves open. The submittal states that the valves will not be reopened. If the valves have to be reopened, then they need to be evaluated for pressure locking and thermal binding.

SNC RESPONSE

NRC letter dated October 16, 1995, "Reclassification of Generic Letter 89-10 Motor-Operated Valve Active Safety Functions - Edwin I. Hatch Nuclear Plant, Units 1 and 2," disagreed with the SNC's reclassification of valves E21-F031A&B (core spray minimum flow valves). Section 2.g of the SER states: "These valves have a safety function to open because after closure, these valves may be required to re-open to support a core spray pump restart and prevent pump damage due to deadheading."

By letter dated October 21, 1996, SNC provided a response to the October 16th SER. The CS minimum flow valves were evaluated for both open and closed active safety functions. However, as stated in the GL 95-07 evaluation sheets, the valves are not required to re-open after closing because the injection flow path would have been established and will remain available even after pump shutdown. The issue of deadhead as stated in SER Section 2.g is therefore not a concern.

NRC QUESTION 100

The Southern Company responses dated 4/17/97 and 3/9/98 to the NRC RAIs do not contain sufficient information to estimate control room χ/Qs using ARCON96. Figure 40-1 in the 3/9/98 letter (Figure 1 in the earlier letter) does not have elevations for the reactor vent (RV1 and RV2) or main stack release points, and the figure can't be used to determine directions and distances from the intakes to the release points. Therefore the following information is requested.

a. A table giving the elevation of each release point and intake

b. A table giving the direction and distance from each intake to each release point

c. One or more figures to scale showing the Hatch Plant layout in sufficient detail to make it possible to estimated directions and distances. The figure should include a distance scale and a indication of North (either plant or true). If plant north is shown, the relationship between plant north and true north should be provided.

d. A figure or map showing the Hatch Plant in its regional setting. The figure should provide some indication of topography.

SNC RESPONSE

Response to NRC Question 100.a

The elevation of each release point and intake considered in the analysis is provided in Table 100.a-1 below:

TABLE 100.a-1

ELEVATIONS OF POINTS CONSIDERED

Point	Elevation (m) Above Grade
RV 1	49.7
RV2	49.7
T1	0 (assumed ground-level release)
T2	0 (assumed ground-level release)
R1	35.2
SL1	18.4
SL2	16.8
Main stack height	120.1
MCR air intake	18.4
TSC intake	3.1

Response to NRC Question 100.b

Table 100.b-1 below provides the direction and distance from the MCR intake to each release point.

TABLE 100.b-1

INTAKE AND RELEASE POINT DISTANCE AND DIRECTION

Wind Direction (°)	Release Point	Intake	Distance (m)
90	R1	MCR	59.8
60	RV1	MCR	69.5
98	RV2	MCR	59.4
55	T1	MCR	112.5
125	T2	MCR	106.5
68	SL1	MCR	92.4
125	SL2	MCR	115.3
85	Stack	MCR	259

Response to NRC Question 100.c

A copy of the Edwin I. Hatch Nuclear Plant General Building Site Plan, Drawing No. E-10173, Revision 7 is provided in Enclosure 6.

The relationship between plant north and true north is the same for Plant Hatch.

Response to NRC Question 100.d

The topography of Plant Hatch, including the locations of the primary meteorological tower and plant structures, is shown in Enclosure 7.

NRC QUESTION 101

Table 40-1 of Southern Nuclear's 3/9/98 response states that some of the window directions were adjusted. I assume that these adjustments were to account for flow around structures. It would be useful to know what the adjustments were, and why they were made.

SNC RESPONSE

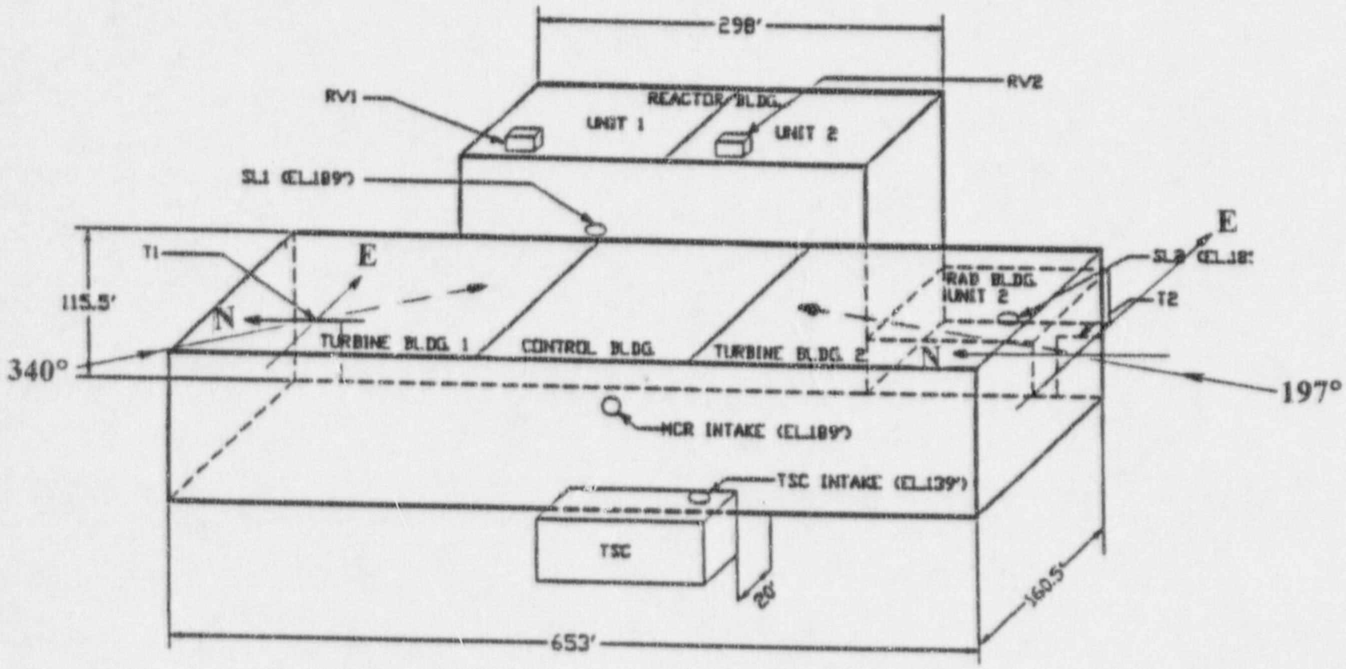
As suggested in ARCON95, the wind direction window sector size was chosen as the direction from the receptor to the source, ± 45°. Based on Figure 40-1 of the March 9, 1998 submittal, the directions from the MCR intake to RV1, RV2, SL1, SL2, T1 and T2 are about 60, 98, 68, 138, 25 and 152 degrees., respectively. If a wind direction window width of ± 45 degrees is applied to these directions, the MCR air intake will not be located downwind from the releases for certain directions.

For example, applying –45° to T1, the wind direction becomes 340° (25° – 45°). However, under this wind direction the release from T1 will not have an impact on the control room air intake because the intake is no longer downwind of T1 (Reference Figure 101-1). Similarly, applying +45° to T2, the wind direction becomes 197° (152° + 45°). Again, the release from T2 cannot have any impact to the intake because the intake is not downwind of T2 . Additionally, T1 and T2 are located at the east side of the turbine buildings, and the air intake is located at the west side of the control building. As a result, if the winds are coming from either 340° or 197°, the releases from T1 and T2 are not expected to have impacts to the air intake.

Therefore, to simulate the impacts properly, the directions to these sources from the intake were adjusted to 55° and 125° for T1 and T2, respectively. Accordingly, the affected wind direction sector size ranges from 10° to 100° and 80° to 170° for T1 and T2, respectively. This way, the wind direction window sector size, as suggested in ARCON95, remains 90°. Based on the same reasoning, the direction from the intake to SL2 was adjusted from 138° to 125°. With these wind direction ranges, releases from T1, T2, and SL2 have the potential to impact on the control room air intake.

FIGURE 101-1

LOCATION OF POTENTIAL RELEASE



LEGEND

T1, T2 Turbine Building Railroad Doors
RV1, RV2 Reactor Building Vents
SL1, SL2 Turbine Building Ventilation System Intake
Ground Level Elevation 129'

NOTES

1. Drawing Not to Scale
2. Elevations Shown are Approximate

NRC QUESTION 102

There are discrepancies between data summaries produced by METQA and those included in 3/9/98 letter response to NRC Questions 38 and 41. Some of these discrepancies may be related to conversions from engineering units used by the HATCH meteorological system to units used in the standard NRC data format. Provide the following relative to the meteorological data system and data processing.

a. What are the original units for the Hatch wind speeds, and delta Ts?

b. There is a large difference in number of occurrences of N wind direction at 10 (all stabilities). Our summary of the data has 334 occurrences compared with 496 in Table 38-8. Were the tables in the letter generated from the same data file that we were given?

c. Figure 41-1 and Table 38-8 don't appear to be consistent. What is wrong?

d. When we have trouble matching the wind directions at the 10 m level, why are we able to match the directions at the 100 m level in Table 38-16?

e. What is the definition of variable winds shown at the bottoms of Tables 38-1 through 38-16? How are these winds distributed in the tables? Why should the wind direction be variable more than 71% of the time at 10 m? Seems much higher than at other locations.

f. What were the precision and units of the delta T values used to determine the stability classes used in compiling the joint frequency tables?

SNC RESPONSE

Response to NRC Question 102.a

The original units for wind speed are miles per hour. Data are recorded to 0.1 mph. The original units for delta temperature are degrees Fahrenheit. Data are recorded to 0.1°F.

Response to NRC Question 102.b

Tables 38-1 through 38-8 in the March 9, 1998 submittal were copied from the 1995 Plant Hatch Annual Meteorological Report transmitted from Pickard, Lowe and Garrick, Inc. to SNC in February 1996. To support this response, the tables were regenerated from our current 1995 data base, and the wind direction sector hourly totals were slightly different than those in the original tables (particularly in the north sector). The regenerated tables, provided in this response as Tables 102.b-1 through 102.b-8, should agree with the NRC data base. No immediate explanation could be found for the discrepancy in the original tables. However, the data base provided to the NRC is correct.

TABLE 102.b-1

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: A

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	3	74	5	0	0	0	82
NNE	11	39	0	0	0	0	50
NE	9	152	3	0	0	0	164
ENE	18	139	6	0	0	0	163
E	11	67	0	0	0	0	78
ESE	16	64	1	0	0	0	81
SE	14	59	5	0	0	0	78
SSE	11	51	6	0	0	0	68
S	5	58	5	0	0	0	68
SSW	7	79	9	0	0	0	95
SW	17	76	5	0	0	0	98
WSW	6	82	7	0	0	0	95
W	11	70	11	0	0	0	92
WNW	10	110	13	0	0	0	133
NW	20	112	12	1	0	0	145
NNW	7	61	9	1	0	0	78
Total	176	1293	97	2	0	0	1568

Periods of Calm (Hours): 0
Variable Direction: 728
Hours of Missing Data: 1

TABLE 102.b-2

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: B

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	7	10	2	0	0	0	19
NNE	5	9	0	0	0	0	14
NE	12	30	0	0	0	0	42
ENE	11	29	0	0	0	0	40
E	12	8	0	0	0	0	20
ESE	4	7	0	0	0	0	11
SE	8	10	2	0	0	0	20
SSE	10	15	2	0	0	0	27
S	4	11	0	0	0	0	15
SSW	9	20	4	0	0	0	33
SW	5	17	0	0	0	0	22
WSW	8	17	0	0	0	0	25
W	2	15	0	0	0	0	17
WNW	3	18	6	0	0	0	27
NW	10	22	1	0	0	0	33
NNW	4	15	1	0	0	0	20
Total	114	253	18	0	0	0	385

Periods of Calm (Hours): 0
Variable Direction: 259
Hours of Missing Data: 1

TABLE 102.b-3

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: C

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	6	13	3	0	0	0	22
NNE	7	5	0	0	0	0	12
NE	11	27	0	0	0	0	38
ENE	12	23	0	0	0	0	35
E	10	11	0	0	0	0	21
ESE	7	6	0	0	0	0	13
SE	8	10	0	0	0	0	18
SSE	8	8	0	0	0	0	16
S	15	12	0	0	0	0	27
SSW	10	16	1	0	0	0	27
SW	18	21	1	0	0	0	40
WSW	5	9	0	0	0	0	14
W	9	15	2	0	0	0	26
WNW	11	21	4	0	0	0	36
NW	10	19	2	0	0	0	31
NNW	16	12	0	0	0	0	28
Total	163	228	13	0	0	0	404

Periods of Calm (Hours): 0
Variable Direction: 301
Hours of Missing Data: 1

TABLE 102.b-4

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: D

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	37	54	3	0	0	0	94
NNE	46	44	1	0	0	0	91
NE	93	146	3	0	0	0	242
ENE	75	95	2	0	0	0	172
E	65	30	0	0	0	0	95
ESE	47	37	0	0	0	0	84
SE	46	50	5	0	0	0	101
SSE	45	37	1	0	0	0	83
S	44	40	9	0	0	0	93
SSW	49	51	7	0	0	0	107
SW	46	51	2	0	0	0	99
WSW	40	30	1	0	0	0	71
W	51	43	6	0	0	0	100
WNW	41	50	17	0	0	0	108
NW	42	67	6	0	0	0	115
NNW	35	81	6	0	0	0	122
Total	802	906	69	0	0	0	1777

Periods of Calm (Hours): 0
Variable Direction: 1323
Hours of Missing Data: 1

TABLE 102.b-5

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: E

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	42	29	0	0	0	0	71
NNE	43	14	0	0	0	0	57
NE	149	97	0	0	0	0	246
ENE	154	83	2	0	0	0	239
E	110	26	0	0	0	0	136
ESE	116	29	1	0	0	0	146
SE	129	62	5	0	0	0	196
SSE	100	62	6	0	0	0	168
S	117	73	14	1	0	0	205
SSW	142	80	9	0	0	0	231
SW	150	39	1	0	0	0	190
WSW	109	51	0	0	0	0	160
W	85	44	2	0	0	0	131
WNW	102	66	5	0	0	0	173
NW	99	64	1	0	0	0	164
NNW	75	51	1	0	0	0	127
Total	1722	870	47	1	0	0	2640

Periods of Calm (Hours): 0
Variable Direction: 2113
Hours of Missing Data: 1

TABLE 102.b-6

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: F

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	23	6	0	0	0	0	29
NNE	22	1	0	0	0	0	23
NE	71	5	0	0	0	0	76
ENE	89	3	0	0	0	0	92
E	59	0	0	0	0	0	59
ESE	65	0	0	0	0	0	65
SE	44	3	0	0	0	0	47
SSE	29	1	0	0	0	0	30
S	66	4	0	0	0	0	70
SSW	68	3	0	0	0	0	71
SW	119	5	0	0	0	0	124
WSW	99	1	0	0	0	0	100
W	100	1	0	0	0	0	101
WNW	55	3	0	0	0	0	58
NW	53	2	0	0	0	0	55
NNW	34	8	0	0	0	0	42
Total	996	46	0	0	0	0	1042

Periods of Calm (Hours): 0
Variable Direction: 835
Hours of Missing Data: 1

TABLE 102.b-7

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: G

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	17	0	0	0	0	0	17
NNE	23	0	0	0	0	0	23
NE	41	0	0	0	0	0	41
ENE	28	0	0	0	0	0	28
E	29	1	0	0	0	0	30
ESE	32	0	0	0	0	0	32
SE	34	1	0	0	0	0	35
SSE	35	0	0	0	0	0	35
S	92	1	0	0	0	0	93
SSW	123	0	0	0	0	0	123
SW	152	0	0	0	0	0	152
WSW	132	0	0	0	0	0	132
W	101	0	0	0	0	0	101
WNW	45	0	0	0	0	0	45
NW	34	0	0	0	0	0	34
NNW	22	0	0	0	0	0	22
Total	940	3	0	0	0	0	943

Periods of Calm (Hours): 0
Variable Direction: 561
Hours of Missing Data: 1

TABLE 102.b-8

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: All

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	135	186	13	0	0	0	334
NNE	157	112	1	0	0	0	270
NE	386	457	6	0	0	0	849
ENE	387	372	10	0	0	0	769
E	296	143	0	0	0	0	439
ESE	287	143	2	0	0	0	432
SE	283	195	17	0	0	0	495
SSE	238	174	15	0	0	0	427
S	343	199	28	1	0	0	571
SSW	408	249	30	0	0	0	687
SW	507	209	9	0	0	0	725
WSW	399	190	8	0	0	0	597
W	359	188	21	0	0	0	568
WNW	267	268	45	0	0	0	580
NW	268	286	22	1	0	0	577
NNW	193	228	17	1	0	0	439
Total	4913	3599	244	3	0	0	8759

Periods of Calm (Hours): 0
Variable Direction: 6120
Hours of Missing Data: 1

Response to NRC Question 102.c

Figure 41-1 and Table 38-8 do not agree for the reason given in response to Question 102.b above. Figure 41-1 was generated on January 22, 1998 from the same data base provided to the NRC. The differences in the percentage of each wind direction sector are quite minor as shown in Table 102.c-1.

TABLE 102.c-1

WIND DIRECTION SECTORS

Direction Sector	Hours in New Table	Hours in Old Table	Percent Direction Change Based on Total Hours
N	334	496	1.8
NNE	270	338	0.7
NE	849	783	0.8
ENE	769	711	0.7
E	439	426	0.2
ESE	432	413	0.2
SE	495	487	0.1
SSE	427	408	0.2
S	571	555	0.2
SSW	687	680	0.1
SW	725	715	0.2
WSW	597	591	0.1
W	568	555	0.2
WNW	580	582	0.0
NW	577	577	0.0
NNW	439	442	0.0

Response to NRC Question 102.d

The data in Tables 38-9 through 38-16 did not change when these tables were regenerated from the current 1995 database. The only discrepancy between the previously provided tables those recently regenerated appears to have been with the 10 m wind direction. However, the largest difference in percent direction change on an annual basis is only 1.8% as shown in Table 102.c-1 above.

Response to NRC Question 102.e

Variable winds in the Plant Hatch data base are defined as hours when the direction varied by more than 75 degrees. No special processing is done for hours with variable winds. They are all included in the normal data base. The high percentage can be attributed to the many hours with low wind speeds at the Plant Hatch 10 m level. The overall average 10 m wind speed for 1995 was 3.5 mph with only about 3% of the hourly average wind speeds above 7.5 mph.

Response to NRC Question 102.f

The precision and stability values used to generate the joint frequency tables were as follows:

Stability Class	°F/100 FT
A	$<$ - 1.0424
B	- 1.0424 $<$ DT/DZ $\leq$ - 0.9333
C	- 0.9333 $<$ DT/DZ $\leq$ - 0.8230
D	- 0.8230 $<$ DT/DZ $\leq$ - 0.2740
E	- 0.2740 $<$ DT/DZ $\leq$ 0.8230
F	0.8230 $<$ DT/DZ $\leq$ 2.1950
G	2.1950 $<$ DT/DZ

NRC QUESTION 103

Were fumigation estimates made? If not, what was the basis?

SNC RESPONSE

As discussed in Section 3.0 of SNC's April 17, 1997 submittal, the χ/Q values used for the site boundary and low population zone (LPZ) are those reported in the FSAR and approved by the NRC. Fumigation was considered in the offsite χ/Q estimates in the original analysis, and no new calculations were performed.

NRC QUESTION 104

For the one hour period the calculations were direction independent. For longer time periods the χ/Qs were direction dependent. For any particular time period, was the maximum 95% χ/Q chosen in all cases (e.g., the maximum 95% tile χ/Q among the

16 95% tile χ/Qs for the direction dependent case) as the appropriate value to use or was some other methodology used?

SNC RESPONSE

As mentioned in response to Question 103, offsite χ/Q values used for the site boundary and LPZ are those reported in the FSAR and approved by the NRC. The proper 95% χ/Q value was used for all cases. No new calculations were performed, and no other method was used to recalculate the 95% χ/Q value.

NRC QUESTION 105

For stack release calculations, was the maximum offsite χ/Q value at or beyond the EAB/LPZ chosen as appropriate or do the values represent the calculations at the EAB or LPZ distance itself?

SNC RESPONSE

The χ/Q values reported in the FSAR are for the site boundary and LPZ. The Unit 2 values for the stack release reported in the FSAR were higher than the Unit 1 χ/Q values. Hence, conservatively, the dose analyses for extended power uprate for both units were based on the Unit 2 χ/Q values at the site boundary and LPZ. .

NRC QUESTION 106

The offsite χ/Q values were calculated using meteorological data measured in the early 1970s. Data used to calculate the control room χ/Q values were from 1995. Are there any significant differences between these two data sets (e.g., was the meteorological measurements program modified between 1970 and 1995)?

SNC RESPONSE

The 1995 meteorological data was used for the control room habitability study because it was the most recent year completed at the time of the study and a typically representative year. Meteorological data collected onsite showed that for the last 10 years (1987 to 1996) the data are quite consistent each year. For the year 1995, all stability classes

(A-G) were within 2% of the 10-year average. For additional information, refer to the SNC response to NRC Question 41 provided in the SNC submittal dated March 9, 1998.

NRC QUESTION 107

What is the change to the operating crew response time due to the Operator Action 'Depressurize with inadequate high pressure injection (Non-ATWS)?

SNC RESPONSE

The scenario described below was originally used in the evaluation of the Human Reliability Analysis (HRA) failure probability associated with the one operator action to be changed as a result of extended power uprate (Depressurization with inadequate high-pressure injection (Non-ATWS). As per the NRC/NRR reviewer request, this scenario was conducted in the Plant Hatch Simulator with two independent licensed operating crews. The scenario was conducted at the current licensed power level of 2558 MWt and at the proposed extended power uprate level of 2763 MWt to provide an assessment of the operator response time as a result of the increase in power. The responses from the operation crews showed no differences or adjustments due to the increased power level nor were there any problems noted in depressurizing the reactor which was the main focus of the scenario. The scenario printouts of the parameters of interest are provided Figure 107-1 (2763 MWt) and Figure 107-2 (2558 MWt). It can be seen from these figures that the operator initiated depressurization at approximately 150 sec into the event for both power levels.

SCENARIO

Initial Conditions

Reactor at full power
PRA-001 = 2763 MWt
PRA-001 = 2558 MWt

Preceding Events

- HPCI is unavailable.
- A loss of Condensate Pumps occurs at T = 60 sec.

- A liquid line break of 0.1898 ft^2 occurs at T = 60 sec resulting in a reactor scram on high drywell pressure in several seconds.
- Diesel generators and RHR and core spray pumps start.
- Operators manually perform initial scram actions, feedwater actions, and pressure control actions per emergency operating procedures (EOPs).
- When generator output is below 80 GMWe, EOP main turbine trip actions are performed.
- Crew members recognize drywell pressure is increasing rapidly, while reactor water level is decreasing rapidly even with RCIC injecting at full capacity.

<u>Required Action Summary</u>

With reactor water level decreasing rapidly, the crew must recognize that reactor water level cannot be maintained above top of the active fuel (TAF) and manually initiate the automatic depressurization system (ADS) and recover reactor water level with the RHR and core spray pumps. As water level is restored, the crew determines that actual reactor water level cannot be determined and floods the vessel. Reactor pressure is established at 50 psid above torus pressure to establish adequate core cooling.

FIGURE 107-1

SIMULATOR SCENERIO AT 2763 MWt

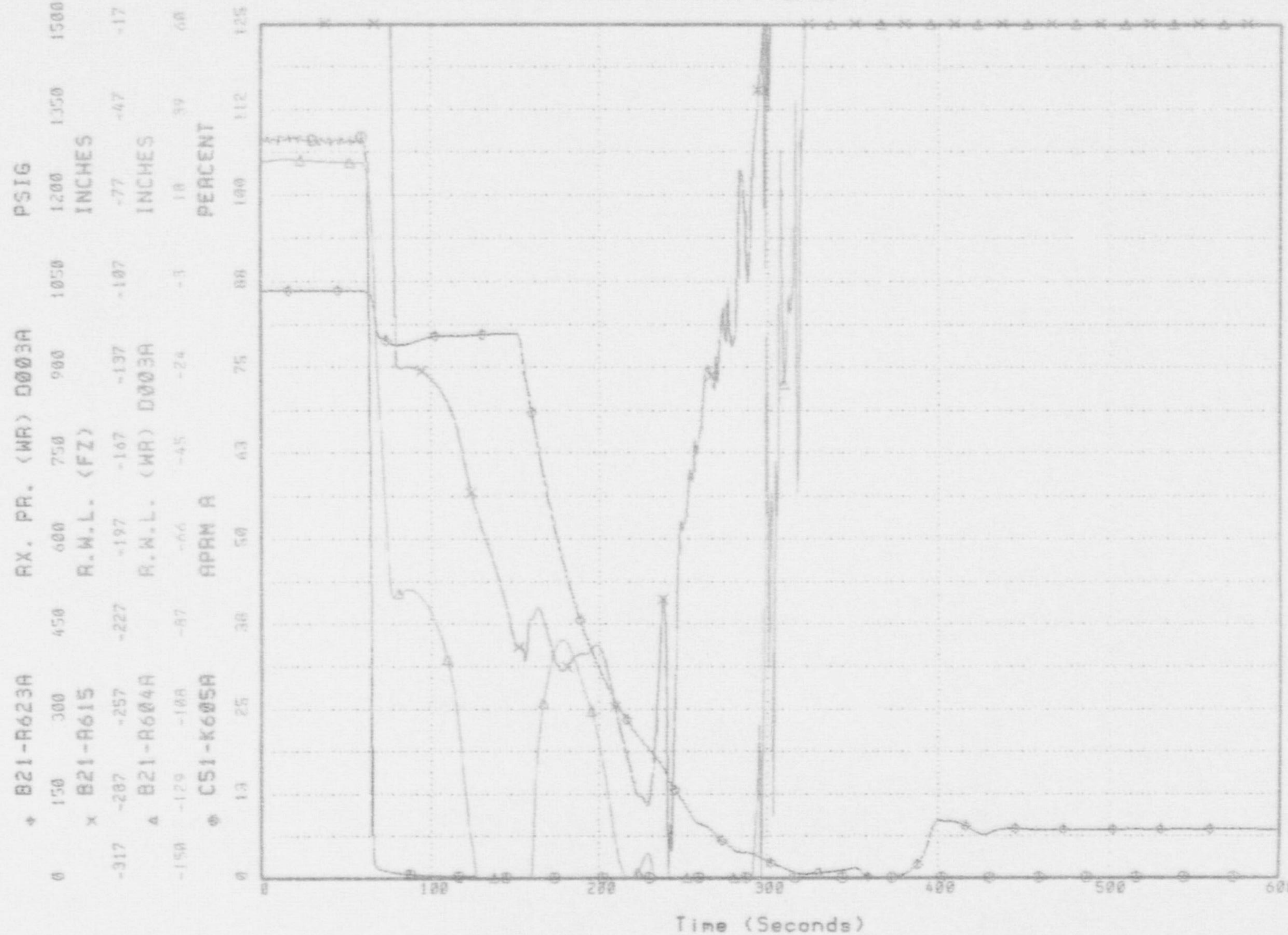
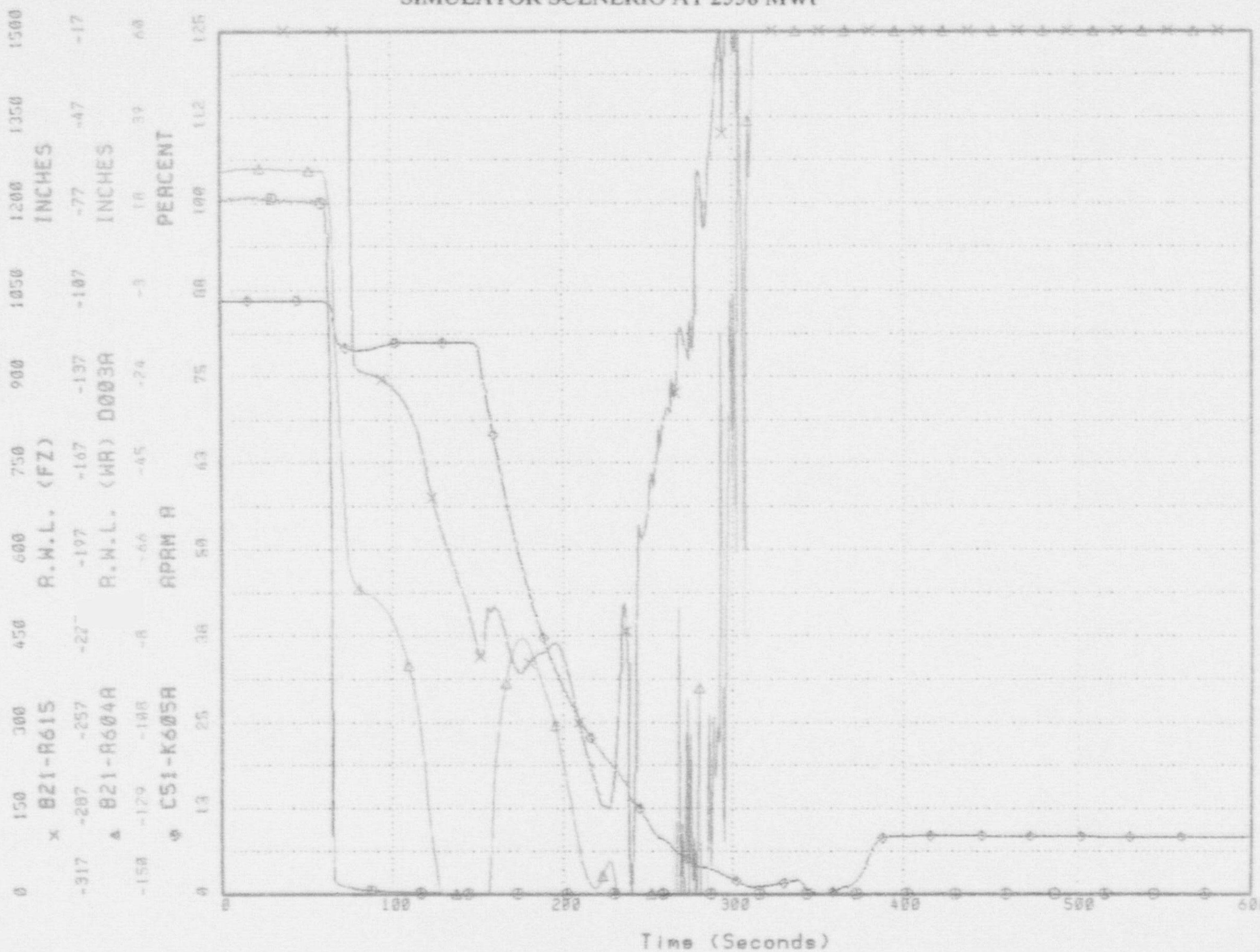


FIGURE 107-2

SIMULATOR SCENERIO AT 2558 MWt



NRC QUESTION 108

Were the concerns discussed in NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" considered with respect to the impact of the proposed extended power uprate?

SNC RESPONSE

SNC is currently addressing the concerns of GL 96-06 via correspondences associated with the generic letter. The evaluation in response to the generic letter will include the impact of the proposed extended power uprate.

NRC QUESTION 109

Pages E3-2 and E3-7 of SNC's August 8, 1997 extended power uprate licensing submittal provide conflicting information regarding circulating water system flow and cooling tower drift. Page E3-2 indicates that there is no increase in the circulating water system flow due to extended power uprate; however, Page E3-7 indicates that circulating water system flow will increase as well as cooling tower drift. Please clarify.

SNC RESPONSE

The circulating water system flow does not change due to extended power uprate and consequently, cooling tower drift also does not change. There is no increase in the circulating water system flow as a result of the proposed extended power uprate. A revision of Page E3-7 of the August 8, 1997 submittal is provided in Enclosure 8 to indicate that cooling tower drift does not change and that the conclusions of the FES relative to cooling tower drift remain valid for extended power uprate.

NRC QUESTION 110

What is the total change in river water temperature, after mixing, with two unit extended power uprate operation?

SNC RESPONSE

Page E3-10 of the SNC submittal dated August 8, 1997 discusses the change in river temperature after complete mixing. The total change in river water temperature, after complete mixing, will not increase significantly (< 0.1°F) as a result of Unit 1 and Unit 2 extended power uprate operation.

NRC QUESTION 111

Please provide a discussion on the potential for cooling tower drift with respect to icing and the impact on trees, other vegetation and roads.

SNC RESPONSE

The FES states that the climate at the site consists of mild, short winters (average monthly minimum temperature of approximately 52°F); therefore, icing conditions are rare, and the probability of icing on nearby roads will be extremely low.

Circulating water flow does not increase due to extended power uprate and consequently, cooling tower drift also does not increase. Since cooling tower drift does not increase, impact of icing on trees, vegetation, and roads does not increase. Based on this information, the conclusions of the FES relative to icing remain valid for extended power uprate.

NRC QUESTION 112

Please provide a mass balance of the water uses for Plant Hatch and the impact of the water consumption as a result of extended power uprate. Mass balance should include a discussion on why the circulating water system blowdown flow will decrease as a result of extended power uprate. And changes in the volume of water loss as evaporation and as drift.

SNC RESPONSE

As discussed in the response to NRC Question 109, circulating water flow does not increase due to proposed extended power uprate and consequently, cooling tower drift also does not increase. Due to an increase in heat load on the cooling towers as a result of extended power uprate, evaporative losses will increase. To compensate for the increase in evaporative losses, cooling tower makeup will be increased slightly and cooling tower blowdown will be decreased by approximately 626 gpm per unit. The net effect of decreasing blowdown flow and slightly increasing makeup to compensate for evaporation losses allows circulating water flow to remain constant. In addition, the cycles of concentration at which the cooling towers operate will not change. The increase in makeup due to the increase in evaporative losses is not significant and is enveloped by values discussed in the FES. As such, the conclusions reached in the FES remain valid for extended power uprate.

Table 112-1 below provides the flow balance of water uses for Plant Hatch.

TABLE 112-1

ENVIRONMENTAL-RELATED OPERATING PARAMETERS

Parameter	**Extended Power Uprate Value**
River Water Withdrawal	22,550 gpm/unit (45100 gpm/2 units)
Cooling Tower Makeup	22,550 gpm/unit
Cooling Tower Evaporation	11,724 gpm/unit
Cooling Tower Drift	1132 gpm/unit (0.2% of flow)
Consumptive Water Use	12,856 gpm/unit
Cooling Tower Blowdown	9694 gpm/unit
Discharge Flow Rate	9694 gpm/unit (19,388 gpm/2 units)

NRC QUESTION 113

Pages E3-35 and E3-36 of SNC's August 8, 1997 extended power uprate licensing submittal incorrectly reference an NRC assessment relative to higher burn-up fuel cycles (Reference 5). Provide the correct reference.

SNC RESPONSE

Southern Nuclear Operating Company has confirmed that there was a typographical error in the Federal Register reference. The correct reference should be as follows: "Federal Register, Volume 53, Number 39, pages 6040-6043, dated February 28, 1988."

NRC QUESTION 114

Page E3-7, section 6.2.1 of SNC's August 8, 1997 extended power uprate licensing submittal states the FES reference of the minimal quantity of groundwater that will be withdrawn for two unit operation. Page E3-7 also states that a permit issued by the state of Georgia Department of Natural Resources imposes limits for withdrawals significantly above the 327 gpm withdrawal rate associated with two unit operation. What is the typical average groundwater withdrawal at current power levels and what is the expected increase as a result of proposed extended power uprate? What is the withdrawal rate allowed by the state of Georgia?

SNC RESPONSE

The FES states that a minimal quantity of groundwater (327 gpm) will be withdrawn from two wells for normal two unit operation. The FES concluded groundwater use at the site is not expected to significantly impact the regional aquifer and is not expected to affect offsite use.

The Plant Hatch Groundwater Withdrawal Permit authorizes withdrawal of 1.1 million gallons per day (gpd) monthly average, and 0.550 million gpd annual average from four wells in the regional aquifer for normal two unit operation. The typical ground water withdrawal rates for two unit operation are 0.167 million gpd (116 gpm), with a maximum value of 0.281 million gpd (195 gpm). These values are significantly lower than the values evaluated in the FES. No change in groundwater withdrawal required to support two unit operation will result from extended power uprate. Based on this information, the conclusions of the FES relative to groundwater use remain valid for extended power uprate.

NRC QUESTION 115

There is no a section in the August 8, 1997 extended power uprate licensing submittal that discusses the chemical discharge. What is the effect of proposed extended power uprate on the biocidal treatment program and the changes to chemical discharges to the river?

SNC RESPONSE

No changes in the cooling tower chemistry program will result from extended power uprate. Circulating water flow does not increase due to extended power uprate and consequently, the cycles of concentration are not changed. Cooling tower blowdown decreases by approximately 626 gpm per unit. The decrease in cooling tower blowdown is compensated by an increase in cooling tower makeup to account for increased evaporation. The net effect of decreasing blowdown flow and slightly increasing makeup to compensate for evaporation losses allows circulating water flow to remain constant. Since the cycles of concentration are unchanged and the cooling tower blowdown rate decreases on each unit, chemical discharges from two unit operation will be reduced slightly. Based on this information, the conclusions of the FES relative to chemical discharges remain valid for proposed extended power uprate.

REFERENCES

1. General Electric Nuclear Energy report, "Plant Hatch Unit 1 RPV Surveillance Materials Testing and Analysis," GE-NE-B1100691-01R1, San Jose, CA, March 1997.

2. General Electric Nuclear Energy report, "E. I. Hatch Nuclear Power Station, Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis", SASR 90-104, San Jose, CA, May 1991.

ENCLOSURE 5

SCS CALCULATIONS SMNH-97-007 AND SMNH-97-008