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Records Management Branch (Document Control Desk)

SUBJECT: Application for amends to licenses DPR-38,DPR-47 & DPR-55, addressing USQ re NPSH requirements for RBS pumps during post-accident recirculation phase.Marked-up UFSAR pages, reflecting results of NPSH analysis,encl.

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TITLE: GL 97-04 Assurance of Sufficient Net Positive Suction Head For Emerg

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October 2, 1998

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

SUBJECT: Duke Energy Corporation
Docket No(s). 50-269, -270, -287
Oconee Nuclear Station Units 1, 2, and 3
Proposed Amendment to the Facility Operating License Regarding Reactor Building Overpressure to Assure Net Positive Suction Head for the Reactor Building Spray Pumps

On October 7, 1997, the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Reactor Building Heat Removal Pumps." Duke Energy Corporation (Duke) provided Oconee's response to GL 97-04 on January 5, 1998. On August 11, 1998, the staff issued a request for additional information which implied that the licensing basis for the available net positive suction head (NPSHa) for the reactor building spray (RBS) pumps may not permit credit for reactor building overpressure. Following receipt of the NRC's letter dated August 11, 1998, Duke reviewed its position to determine if Oconee Nuclear Station Units 1, 2, and 3 were operating within their licensing basis regarding NPSH for the RBS and LPI pumps. On August 19, 1998, Duke concluded that utilizing a limited RB overpressure to assure adequate NPSH available to the RBS pumps was not within the licensing basis. As a result, this condition was reported to the NRC in accordance with 10 CFR 50.72, an operability evaluation was completed, and a Licensee Event Report (LER) was submitted in accordance with 10 CFR 50.73 on September 17, 1998. In addition, Duke responded to the staff's August 11, 1998, request for additional information in a letter dated September 17, 1998, and stated that a license amendment request (LAR) to address the USQ would be submitted by October 2, 1998.

A076

Pursuant to 10 CFR 50.90, this letter submits a LAR for Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station Units 1, 2, and 3, respectively. This request addresses a USQ pertaining to the NPSH requirements for the RBS pumps during the post-accident recirculation phase. The changes being proposed by Duke allow reactor building overpressure to be credited in the calculation of the available NPSH for the RBS pumps for a limited period of time during the post-accident recirculation phase.

The attached LAR reflects the results of the current RBS NPSH calculation. Duke is currently performing additional analyses to determine if additional margin can be established regarding the RBS NPSH requirements. These additional analyses may require an update to the information presented in the attached amendment request. Even though these additional analyses are ongoing, Duke wanted to submit the available information expeditiously to initiate timely resolution of the USQ.

In addition, further engineering review of the NPSH calculations determined that the initial unthrottled RBS flow during the injection phase had not been evaluated with respect to NPSHa. A Problem Investigation Process (PIP) report was written to address this deficiency and evaluate the NPSH during the initial stages of injection. This issue was promptly evaluated, and determined to be reportable under 10 CFR 50.72 on October 1, 1998. Duke plans to resolve this issue in accordance with 10 CFR 50, Appendix B, Criterion XVI, and it will not impact the information provided in the attached LAR.

This LAR contains the following attachments:

Attachment 1 provides marked-up UFSAR pages that reflect the results of the NPSH analysis.

Attachment 2 provides the description and technical justification of the proposed changes to the Updated Final Safety Analysis Report (UFSAR).

Attachment 3 documents the determination that the amendment contains No Significant Hazards Considerations pursuant to 10 CFR 50.92.

Attachment 4 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement pursuant to 10 CFR 51.22(c)(9).

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this LAR has been previously reviewed and approved by the Oconee Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this LAR is being sent to the State of South Carolina.

Inquiries on this matter should be directed to J. E. Burchfield, Jr. at (864) 885-3292.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Attachments

xc w/attachments:

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AFFIDAVIT

W. R. McCollum, Jr., being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this revision to the Oconee Nuclear Station License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., Site Vice President

Subscribed and sworn to me: 10-2-98
Date

Notary Public: Connie M. Breazeale

My Commission Expires: 2-12-2003
Date

SEAL

Attachment 1

Markup Pages of the UFSAR

The NPSH available to the Low Pressure Injection and Reactor Building Spray pumps during the post-LOCA recirculation phase has been calculated based on:

a. "As Built" piping drawings.

b. Pipe and fitting losses calculated using the information in Crane Technical Paper No. 410.

5 c. Total indicated flow in a single string (i.e., consisting of one Low Pressure Injection Pump and one Reactor Building Spray pump served by a single sump suction line) is 4,000 gal/min. This consists of 3,000 gal/min to the Low Pressure Injection pump and 1,000 gal/min to the Reactor
5 Building Spray pump. Instrument uncertainties have been applied to these values to provide
5 conservatism in the NPSH analysis.

5 d. Sump water temperatures and Reactor Building pressures were determined from analysis of hot
5 leg break with conservative building cooling assumptions (LPSW temperature, RBCU capacity,
5 etc.). RBCU capacity was varied from 60 million Btu/hr to 160 million Btu/hr to produce bounding pressures for all sump temperatures.

e. Reactor Building Spray pump shaft center line ~~are~~ at elevation 760 ft. 1 in.

f. Low Pressure Injection pump shaft center line at elevation 761 ft. 1 in.

5 g. Water level in the Reactor Building sump is ~~782 ft. 6 in~~ at elevation 781.15 ft. based on the following assumptions:
5 (height above Reactor Building basement level is ~~5.0~~ 4.27 ft.)

1) The Technical Specification ~~minimum~~ initial levels were used for the BWST and the CFT's, with
5 six feet of level remaining in the BWST at ~~time~~ completion of switchover to RBES.

2) Some water is maintained in the Reactor Building atmosphere as vapor. The quantity was determined using the results of a ~~CONTEMPT~~ FATHOMS Computer Run for a ~~6.0~~ 14.1 ft² break with 2 ~~fan coolers~~ RBCUs and one Reactor Building Spray pump operating.

3) The break is conservatively assumed to occur at the top of the hot leg, thereby keeping the Reactor Coolant System full.

5 h. Unit 3 NPSH available to each pump in the "B" string, LP-P3B and BS-P3B. Calculations show
5 this to be the worst case flowpath of all possible units and trains.

5 INSERT A

~~The required NPSH's indicated above reflect the manufacturer's certified test.~~

5 Available NPSH has been determined to meet or exceed the required NPSH for worst case accident
5 conditions with conservative inputs as identified above. Curves of total dynamic head and NPSH versus
5 flow are shown in Figure 6-5 for the Reactor Building Spray Pumps and in Figure 6-17 for the Low
5 Pressure Injection Pumps. These curves are representative in nature and are provided for information
5 only. They are not intended to constitute design commitments or performance requirements for the
5 pumps. Refer to the Inservice Test Program for actual performance requirements for BS and LPI pumps.

Bases of Leakage Estimates: While the reactor auxiliary systems involved in the recirculation complex are closed to the Auxiliary Building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

a. Valves

1) Disc leakage when valve is on recirculation system boundary.

2) Stem leakage.

INSERT A

Results of this NPSH analysis are presented in Table 6-33. Note that this table represents a combination of several building response analyses with varying RBCU capacity assumptions. The reactor building pressure at each sump temperature was based upon the RBCU capacity which produced the minimum building pressure for that sump temperature. The results show that overpressure is credited for the Reactor Building Spray Pumps at sump temperatures above approximately 210°F. Approximately 2.2 psi of reactor building overpressure can be credited in the calculation of available NPSH for the RBS pumps from approximately 3,000 seconds to approximately 30,000 seconds post-accident.

NPSH REQUIREMENTS FOR RBS & LPI OPERATION AT ELEVATED SUMP TEMPERATURES

SUMP TEMP (F)	NPSHr RBS (FT)	NPSHr Corrected RBS (FT)	NPSH AVAILABLE RBS (FT)	NPSHr LPI (FT)	NPSHr Corrected LPI (FT)	NPSH AVAILABLE LPI (FT)	NPSH REDUCTION FACTOR (FT)
195	17	16.50	25.98	13	12.50	24.70	0.50
200	17	16.40	24.21	13	12.40	22.93	0.60
205	17	16.30	22.35	13	12.30	21.08	0.70
210	17	16.25	20.46	13	12.25	19.20	0.75
215	17	16.15	19.93	13	12.15	18.67	0.85
220	17	16.10	19.83	13	12.10	18.57	0.90
225	17	16.00	21.35	13	12.00	20.09	1.00
229.6	17	15.90	25.80	13	11.90	24.54	1.10

BUILDING PRESSURE REQUIREMENTS FOR RBS & LPI OPERATION AT ELEVATED SUMP TEMPERATURES

	BUILDING PRESSURE				
SUMP TEMP (F)	REQUIRED RBS (PSIG)	AVAILABLE RBS (PSIG)	REQUIRED LPI (PSIG)	AVAILABLE LPI (PSIG)	Psat (Gauge)
195	-3.46	0.50	-4.61	0.50	-4.31
200	-2.37	0.90	-3.51	0.90	-3.17
205	-1.17	1.35	-2.31	1.35	-1.93
210	0.15	1.90	-0.98	1.90	-0.58
215	1.58	3.15	0.45	3.15	0.89
220	3.16	4.70	2.03	4.70	2.49
225	4.84	7.05	3.71	7.05	4.21
229.6	6.52	10.60	5.40	10.60	5.94

OVERPRESSURE REQUIREMENTS FOR RBS & LPI OPERATION AT ELEVATED SUMP TEMPERATURES

SUMP TEMP (F)	OVERPRESS REQUIRED RBS (PSI)	OVERPRESS AVAILABLE RBS (PSI)	OVERPRESS REQUIRED LPI (PSI)	OVERPRESS AVAILABLE LPI (PSI)	Psat (Gauge)
195	0.00	4.81	0.00	4.81	-4.31
200	0.00	4.07	0.00	4.07	-3.17
205	0.00	3.28	0.00	3.28	-1.93
210	0.15	2.48	0.00	2.48	-0.58
215	0.69	2.26	0.00	2.26	0.89
220	0.67	2.21	0.00	2.21	2.49
225	0.63	2.84	0.00	2.84	4.21
229.6	0.58	4.66	0.00	4.66	5.94

NOTE: Required overpressure is required building pressure less Psat, except where Psat is negative. Where Psat is negative, required overpressure is required building pressure above atmospheric. Available overpressure is available building pressure less Psat.

TABLE 6-33

Attachment 2

Description and Justification of the Proposed Changes to the Oconee Updated Final Safety Analysis Report

Background

On May 5, 1972, Duke filed Amendment No. 31 to its Application for License for the Oconee Nuclear Station. This submittal provided Revision 19 to the Oconee Final Safety Analysis Report (FSAR) which included a discussion of the evaluation used to establish minimum net positive suction head (NPSH) for the low pressure injection (LPI) and reactor building spray (RBS) pumps. The referenced analysis addressed two cases. One case took credit for reactor building (RB) pressure, and the other assumed the RB total pressure was equal to the saturation pressure of the sump water. The referenced analysis demonstrated that the available NPSH (NPSHa) was greater than the required NPSH (NPSHr) for the RBS and LPI pumps for both cases.

On July 6, 1973, the Nuclear Regulatory Commission (NRC) issued a Safety Evaluation for Oconee Nuclear Station, Units 2 and 3. This Safety Evaluation contained the original licensing basis for the NPSHa for the RBS pumps. The Safety Evaluation contained the following statements:

1) in Section 7.1.5, "... the applicant used the Oconee "as-built" configuration, sizes, layouts, etc., and made assumptions based on both credit for reactor building pressure and no credit for reactor building pressure (saturation pressure of sump water);"

2) in Section 7.1.5, "In all cases for the low pressure injection pumps, the available NPSH exceeded the required NPSH for the worst case assumptions of maximum sump temperature and no credit for building pressure;" and

3) in Section 7.2, "The staff requested that the applicant provide analysis in the FSAR to justify that the containment spray pumps have adequate net positive suction head. This analysis was performed with the analysis for the ECCS pumps described in Section 7.1.5 and the results were the same."

The NRC Safety Evaluation for Oconee Nuclear Station Unit 1 and its Supplements are silent regarding the assumptions that were utilized in the NPSH calculations for the LPI and RBS pumps.

Oconee Nuclear Power Station Units 1, 2, and 3 did not commit to meet the recommendations of Regulatory Guide (RG) 1.1.

History of Revised RBS NPSH Calculations

To address potential non-conservatisms identified with the original NPSH calculations for the RBS, a new NPSH calculation (OSC-4361) was developed in May 1991. This calculation credited a limited amount of reactor building overpressure to assure the available NPSH for the RBS pumps. The calculation included the following statement: "The RB pressure is 2 psig or greater for sump temperatures below 212 F. Although Oconee did not take credit for RB pressure in its original NPSH analysis, Oconee is exempted from Regulatory Guide 1.1 and thus is allowed to do so." The calculation contained a reference to Section 7.1.5 of the NRC's Safety Evaluation for Units 2 and 3 dated July 6, 1973.

In February 1992, another calculation (OSC-4467) was developed which determined the needed reactor building pressure for a range of expected post-accident sump temperatures.

In April 1996, revision 1 to OSC-4467 was issued, and the UFSAR was updated as a result of the calculation. The supporting 10 CFR 50.59 evaluation concluded that the changes to the UFSAR resulting from the revised NPSH analysis did not involve an unreviewed safety question.

Historically, Duke believed its licensing basis only required the ability to demonstrate that the NPSHa for the RBS pumps was greater than the NPSHr. Duke did not believe it was prohibited from crediting reactor building overpressure, as long as the RB overpressure required to assure adequate NPSH for the RBS pumps was less than the

minimum predicted RB pressure based on conservative analysis. The rationale for this belief was:

1) Duke believed that the precedent had been set to utilize RB overpressure if needed to assure adequate NPSH. This belief was based on the fact that the original UFSAR contained a summary of a NPSH analysis which addressed cases with and without RB overpressure assumed. This analysis was referenced in the NRC's Safety Evaluation dated July 6, 1973; and

2) Oconee Units 1, 2, and 3 were not licensed to Regulatory Guide 1.1.

On October 7, 1997, the NRC issued GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Reactor building Heat Removal Pumps." It requested licensees to submit information necessary to confirm the adequacy of the NPSHa for emergency core cooling and reactor building spray pumps.

Duke provided Oconee's response to GL 97-04 on January 5, 1998. In the response, Duke stated:

1) "the current design-basis NPSH analysis is different than that reviewed and approved by the NRC;"

2) "The original SER does acknowledge the crediting of overpressure, although the current analysis credits somewhat higher building pressure;"

3) "Reactor building overpressure was credited in the calculation of available NPSH;"

4) the RB overpressure required to assure adequate available NPSH was 2.28 psi for the RBS pumps, and 0 psi for the LPI pumps; and

5) the RB overpressure available was 2.90 psi.

This response was based on revision 2 to OSC-4467.

On August 11, 1998, the NRC staff issued a request for additional information (RAI) which implied that the licensing basis for the available NPSH for the RBS pumps did not permit RB overpressure to be credited. Following receipt of the NRC's letter dated August 11, 1998, Duke reviewed its position to determine if Oconee Nuclear Station Units 1, 2, and 3 were operating within their licensing basis regarding NPSH for the RBS and LPI pumps. On August 19, 1998, Duke concluded that utilizing a limited RB overpressure to assure adequate NPSH available to the RBS pumps was not within the licensing basis. As a result, this condition was reported to the NRC in accordance with 10 CFR 50.72, an operability evaluation was completed and a Licensee Event Report (LER) was submitted in accordance with 10 CFR 50.73 on September 17, 1998.

Additionally, Duke responded to the NRC's RAI on September 17, 1998. The LER and Duke's response to the NRC's RAI committed to submit a license amendment request to resolve the USQ by allowing RB overpressure to be credited in the calculation of the NPSHa for the RBS pumps.

Description of the Proposed Changes

Revision 6 to OSC-4467 concludes that approximately 0.7 psi of reactor building overpressure is required to assure adequate available NPSH for the RBS pumps for a limited period of time during the post-accident recirculation phase. This is determined from the Table titled "Overpressure Requirements for RBS & LPI Operation at Elevated Sump Temperatures" contained in Table 1 of Attachment 2.

Reactor building overpressure is required to be credited from approximately 3000 seconds (i.e., the time that pump suction is transferred to the sump) to approximately 27,000 seconds (i.e., the time the sump temperature drops below 210°F) following the onset of the LOCA. Figure 3 of Attachment 2 provides a graph of sump temperature versus time.

To avoid the need for future license amendment requests due to small changes in the RBS NPSH analysis, Duke is requesting additional licensing margin. Duke is requesting

the ability to credit approximately 2.2 psi of containment overpressure from approximately 3000 seconds to approximately 30,000 seconds following the onset of the LOCA to meet the NPSH needs for the RBS pumps. The Table titled "Overpressure Requirements for RBS & LPI Operation at Elevated Sump Temperatures" contained in Table 1 of Attachment 2 provides a column which identifies the available overpressure at the elevated temperatures. Approximately 2.2 psi is the minimum available reactor building overpressure for the sump temperatures of concern.

Changes to the UFSAR

To reflect the revised NPSH analysis, several changes to Section 6.1.3 of the UFSAR are necessary.

Section 6.1.3 of the UFSAR contains a list of factors that were considered in the calculation of available NPSH for the LPI and RBS pumps.

Item "d" of this list states: "Sump water temperatures and Reactor Building pressures were determined from analysis of hot leg break with conservative building cooling assumptions (LPSW temperature, RBCU capacity, etc.)." The following is proposed to be added to this statement: "RBCU capacity was varied from 60 million BTU/hr to 160 million BTU/hr to produce bounding pressures for all sump temperatures."

Item "e" states: "...shaft center line are elevation...". This is proposed to be revised to state: "...shaft center line at elevation...". This is an editorial correction.

Item "g" states "Water level in the Reactor Building sump is 782 ft. 6 in. Based on the following assumptions: (height above Reactor Building basement level is 5.0 ft.)" This statement is proposed to be changed as follows: "Water level in the Reactor Building sump is at elevation 781.15 ft. Based on the following assumptions: (height above Reactor Building basement level is 4.27 ft)." Voluntary Licensee Event Report 269/97-10, dated January 8, 1998, documented issues associated with the sump water level calculation. The resultant calculation formed the basis for this change.

Item g.1) states "The Technical Specification minimum levels were used for the BWST and the CFT's, with six feet of level remaining in the BWST at time of switchover." This is proposed to be revised as follows: "The Technical Specification minimum initial levels were used for the BWST and the CFT's, with six feet of level remaining in the BWST at completion of switchover to RBES."

Item g.2) states: "Some water is maintained in the Reactor Building atmosphere as a vapor. The quantity was determined using the results of a CONTEMPT Computer Run for a 5.0 ft^2 break with 2 fan coolers and one Reactor Building Spray pump operating." This statement is proposed to be revised to state: "The quantity was determined using the results of a FATHOMS Computer Run for a 14.1 ft^2 break with 2 RBCUs...".

The following statement and referenced table are proposed to be added to the UFSAR: "Results of this NPSH analysis are presented in Table 6-33. Note that this table represents a combination of several building response analyses with varying RBCU capacity assumptions. The reactor building pressure at each sump temperature was based upon the RBCU capacity which produced the minimum building pressure for that sump temperature. The results show that overpressure is credited for the Reactor Building Spray Pumps at sump temperatures above approximately 210°F. Approximately 2.2 psi of reactor building overpressure can be credited in the calculation of available NPSH for the RBS pumps from approximately 3,000 seconds to approximately 30,000 seconds post-accident."

The UFSAR contains the statement: "The required NPSH's indicated above reflect the manufacturer's certified test." This statement is proposed to be deleted, because the revised NPSH analysis relies on additional data provided by the vendor.

Attachment 1 contains a mark-up of the affected UFSAR pages.

Justification of the Proposed Changes

Introduction

The RBS system provides cooling of the reactor building atmosphere to limit post-accident temperatures and pressures. It operates in conjunction with the reactor building cooling system and the LPI system to remove sufficient heat from the containment atmosphere to maintain post-accident conditions within the applicable design limits.

Additionally, the RBS system functions to remove iodine from the post-accident reactor building atmosphere. Credit is taken for this capability in the analysis of the Maximum Hypothetical Accident (MHA). The MHA involves a gross release of fission products; however, no mechanism whereby such a release occurs is postulated, because a multitude of failures of engineered safeguards would have to occur. No credit is taken for the capability of the RBS system to remove iodine in the current Loss of Coolant Accident (LOCA) analyses.

The RBS system consists of two redundant trains, each including a RBS pump with associated valves and piping, and an array of spray nozzles located on the spray headers in the upper containment. The RBS system delivers borated water through the spray nozzles to the reactor building atmosphere. The spray condenses the steam produced by a steam line break or LOCA, thereby reducing the reactor building temperature and pressure. Prior to the reactor building pressure reaching the limit of Technical Specification 3.5.3 (i.e., ≤ 15 psig), the engineered safeguards system initiates a signal which automatically starts the RBS pumps and opens the spray header reactor building isolation valves. The RBS pumps take suction from the LPI system headers upstream of the LPI pumps.

During the injection mode, the LPI suction headers are aligned to the borated water storage tank (BWST). This alignment is maintained until the BWST is depleted. After the BWST volume has been depleted, the LPI suction headers are aligned to take suction from the reactor building

emergency sump. This allows the spilled coolant and spray water to be recirculated.

Reactor Building Response Analysis

An analysis was performed to determine the reactor building response to a large hot-leg break LOCA. This break location has been previously identified as the limiting scenario for worst-case conditions to determine the NPSHa for the RBS pumps. Due to the termination of steaming from the break, the reactor coolant system (RCS) energy enters the reactor building in the form of warm water rather than steam. Therefore, the RB pressure is minimized with respect to other break locations, with the sump temperature being maximized. These two parameters directly affect the NPSHa for the RBS pumps. The combination of low RB pressure and high sump temperature is worst-case for NPSHa determination.

The FATHOMS/DUKE-RS code (FATHOMS) was utilized for the analysis. The methodology utilized in the analysis is consistent with that given in the Duke Power topical report DPC-NE-3003-PA. Note: the designator at the end of the topical report number defines whether the topical report was provided to the NRC for review (i.e., "P"), or approved by the NRC (i.e., "PA"). The Safety Evaluation Report for this topical was issued by the NRC on March 15, 1995. In the Summary and Conclusions section of the SER, the following statement is found:

"The DPC-NE-3003-P basic methodology discussed in this evaluation, with appropriate adjustments to reflect potential plant modifications, may be used by the licensee to perform future re-analyses in support of licensing applications related to reactor building accident response."

The FATHOMS model used to determine the reactor building pressure and temperature response for use in the calculation of NPSHa for the RBS pumps is identical to the model described in DPC-NE-3003-PA. Section 2.4.2 of this topical report gives an extended description of the simulation model used for reactor building analysis with the FATHOMS code. There are different assumptions made for various initial and boundary conditions to provide a conservative answer (low

building pressure, high sump water temperature) for the NPSHa determination. There were no restrictions or requirements placed on the FATHOMS reactor building analysis model by the NRC.

There is no section within the ANSI/ANS-56.4-1983 standard explicitly dealing with reactor building response calculations for NPSHa determination. However, the following statement is found in section 4.1 of the standard:

"All considerations regarding modeling philosophy, assumptions, and input, as well as those relating to the spectrum of cases and events addressed, whether discussed herein or not, shall be handled in a manner which will render each analysis conservative for its stated purpose."

In the selection of the initial and boundary conditions for the analyses, consideration was given as to which direction the initial value of each parameter would affect the analysis results (reactor building P-T response). In some cases, the selection of the maximum initial value for a parameter would increase the sump temperature following the LOCA, but would also increase the RB pressure. In these cases, a judgment was made as to which direction the parameter value should be selected (minimum or maximum value) to have the maximum (worst-case) effect on the result with respect to the NPSHa determination. Physical restrictions and plant operating history were also considered.

Direct comparisons between the assumptions made in the minimum containment pressure analysis conducted by Framatome to provide the minimum backpressure for peak clad temperature determination (discussed in the Oconee UFSAR, Chapter 15.14.3) and the assumptions made in this NPSHa analysis are not entirely appropriate. In the minimum containment pressure analysis, initial values for each parameter of interest are selected with only one factor in mind: minimize the post-LOCA RB pressure. In the NPSHa analysis, it is desired to minimize the post-LOCA RB pressure and maximize the post-LOCA sump temperature at the same time. As discussed previously, this leads to differing assumptions between the two analyses.

It was determined in the analyses that high Low Pressure Service Water (LPSW) temperatures are conservative for the NPSHa analyses. This minimizes the capacity of the Reactor Building Cooling Units (RBCUs) and Low Pressure Injection (LPI) coolers to remove energy from the Reactor Building following the LOCA. It was also determined that the degree of fouling of the RBCUs changed the reactor building response in such a way that it was not possible to assume a certain level of fouling for a certain case, and to distinguish this particular case as the most limiting. At different sump temperatures, the RB pressure reached lower levels as the amount of RBCU fouling assumed was decreased. Therefore, a set of cases is analyzed with the FATHOMS code with the degree of RBCU fouling changed in each case.

As discussed previously, the initial values of some initial conditions were chosen to minimize the post-LOCA RB pressure and maximize the post-LOCA sump temperature at the same time. For some variables, a sensitivity case was performed to ensure that the proper initial conditions had been chosen to accomplish this. For instance in one case, the initial RB pressure was lowered to -2.45 psig, with an initial RB temperature of 80°F. The results of this case demonstrated that an initial pressure this low does not produce the limiting RB response for NPSHa determination, when the physical process required to arrive at an initial pressure this low is considered.

The values assumed in the limiting case for several key parameters are listed below:

- Initial reactor building pressure of 13.7 psia
- Initial reactor building temperature of 160°F. This is a conservatively high value, which assumes the auxiliary reactor building coolers are not in service.
- Reactor building free volume (CFV): 1,897,900 ft^3 (Nominal + 1%)
- Passive heat sink areas: Nominal - 1%
- BWST initial temperature: 100°F
- BWST initial level: 44.3 ft (Technical Specification minimum, minus uncertainty)

- Sump re-circulation switchover point on BWST level: 7.68 ft (maximum, plus uncertainty)
- RBCU capacity: 60 E6 Btu/hr, at design conditions (combined capacity of two operating RBCUs)
- LPI cooler capacity: 93 E6 Btu/hr, at design conditions (represents maximum tube plugging level permitted, plus potential fouling during cycle length)
- LPSW temperature: 85°F
- RBS flow: 1085 gpm (nominal value of 1000 gpm + uncertainty - flow spilled out open RBS drain line) - The uncertainty is added, rather than subtracted, to match the assumption in the $NPSH_r$ analysis (OSC-4467). (The flow spillage assumption represents water which was pumped by the RBS pump, but did not pass through the spray nozzles due to the open drain lines)
- Single failure assumed: Loss of 4160 V switchgear, eliminates one RBCU and one train of HPI/LPI/RBS (this minimizes ability of ESF equipment to remove energy from RB)
- Delay time for LPSW flow to reach LPI coolers following switchover: 30 minutes. This is a conservatively long delay time that reflects operator action to valve in LPSW flow to the LPI coolers following the swapover to sump recirculation.
- Decay heat model: ANS 5.1-1979 + 2 sigma uncertainty (maximizes energy to RB)

The building pressure, vapor temperature, and sump temperature responses as a function of time are shown in Figures 1, 2, and 3 as a representative case which reflect the above assumptions. Additional cases were run varying the RBCU capacity between 60 E6 Btu/hr and 160 E6 Btu/hr. These cases show the same general trends as illustrated in these figures and are not represented here. In each figure, the start of sump recirculation mode is visible, as containment cooling decreases at this point. Figure 4 provides a conservative overlay of building pressure versus sump temperature for the spectrum of cases analyzed.

Net Positive Suction Head Analysis

The RBS NPSH analysis utilizes the following design inputs:

1. Worst case initial reactor building pressure of 13.7 psia.
2. A minimum available building overpressure of approximately 2.2 psi. Building overpressure is defined as the difference between the reactor building pressure and the vapor pressure of the sump fluid.
3. A sump temperature range versus time based on the previously described reactor building response analysis.
4. An elevation head determined from as-built station piping drawings and detailed post-LOCA reactor building water level analysis.
5. Friction losses determined from a hydraulic computer program (WOODS) with industry accepted resistance coefficients (from Crane Technical Paper 410) and as-built station piping drawings. The Unit 3B pump suction piping was modeled since it has been shown to introduce the most friction losses of any train and is therefore analytically conservative.
6. Flow rates consistent with accident analysis assumptions and emergency operation procedure guidance, with instrument uncertainties applied in the most conservative direction.
7. Sump screen debris blockage of 50%.
8. NPSH requirements provided by pump manufacturer and adjusted by application of high temperature correction factors.

A spectrum of scenarios were evaluated to ensure the worst case bounding conditions for the NPSH analysis were considered. A LOCA resulting from a large break of a hot leg was determined to be the most limiting when compared to small breaks or a large break of a cold leg. The hot leg break resulted in higher sump temperature and lower reactor building static pressure, both of which tend to be more limiting for NPSH considerations. A single failure of one low pressure injection train, RBS train, and one reactor building cooling unit (RBCU) was assumed. The remaining RBCU capacity was varied over a range of 60 million to 160 million BTU/hr to cover the entire range of temperatures for

which reactor building overpressure is required. Higher cooling capacity assumptions produced lower building pressures. While the amount of overpressure credit required is relatively insensitive to building conditions, lower cooling capacity assumptions produced the highest peak sump temperature, and resulted in the greatest amount of time for the sump to cool to a temperature where overpressure credit is no longer required.

In all cases, the NPSH analysis showed that the available NPSH exceeded the required NPSH for all sump temperatures for the RBS pumps. A visual representation of this conclusion is provided in Figure 4.

Emergency Operating Procedures

The emergency operating procedures (EOP) do not require revision as a result of this change. The analyses assume EOP setpoints and operator actions as currently stated in the EOP. Operator actions to monitor reactor building pressure and its potential impact on RBS pump performance are not necessary. The limited amount and duration of overpressure have been demonstrated as being available based on conservative reactor building analyses. The impact of no overpressure on RBS pump performance resulting from a beyond design basis event that results in a loss of containment integrity is not judged to be significant. Pump performance under no overpressure conditions is addressed below.

Analysis of Failure to Achieve or Loss of Reactor Building Overpressure

If the required reactor building overpressure is not available within the required time frame or a gross failure of the reactor building occurs, a NPSH deficiency for the RBS pumps could result due to the conditions in the reactor building during the post-LOCA sump recirculation mode of operation. This deficiency would be no more than 1.7 feet of NPSH and could exist from approximately 3,000 seconds to approximately 30,000 seconds post-accident. The vendor for the RBS pumps has confirmed that the pumps would continue to function, although in a degraded manner. The RBS pumps are ruggedly constructed and the cavitation effects would not

result in mechanical failure. Under these conditions, the flow would be reduced from 1150 gpm to approximately 1000 gpm.

This flow rate is adequate for the pumps to perform their design basis accident mitigation function. It is expected that these pumps would easily be capable of functioning under these conditions long enough for the sump to cool sufficiently to eliminate postulated NPSH deficiencies associated with a loss of reactor building integrity.

A gross failure of the reactor building is not expected. Duke has a rigorous inservice test program in place to ensure that the reactor building isolation valves will respond when called upon to actuate to their required safety positions. The Appendix J leak testing program ensures that these valves will maintain leakage within acceptable limits once properly positioned. In addition, there are procedural limits and precautions in operating and test procedures which ensure that reactor building integrity is maintained in compliance with the Technical Specification requirements.

Fission Product Scrubbing

For design basis accidents, fission product scrubbing by the RBS system is expected to occur if the reactor building conditions necessitate the use of the RBS. With the reactor building isolated, the post-accident reactor building pressure is seen to be greater than that needed for the NPSH margin. Gross failure of the reactor building integrity is needed to preclude the reactor building backpressure. Such a gross failure is very unlikely and is beyond the design basis.

Impact on Severe Accident Risk

The RBS system performs no function related to maintaining reactor coolant system inventory or core heat removal. Therefore, the core damage frequency results are not affected by the potential failure modes of the RBS system.

Release mitigation following core damage accidents is accomplished by isolating the reactor building via the

engineered safeguards protective system and by preventing reactor building failure from overpressurization by either the reactor building cooling system or the RBS system. For core damage sequences where the RBS system is available, the RBS system also plays a role in scrubbing the fission products in the reactor building atmosphere. In the current PRA (Rev. 2), the RBS system is seen to be failing for reasons other than the NPSH considerations (for example, loss of all power, reactor building bypass, etc.) for core damage sequences involving large early release. No sequences that contribute to the large early release are impacted by a loss of RBS during recirculation.

The RBCUs provide adequate heat removal to prevent reactor building failure as a result of overpressurization. As long as the RBCUs remain in operation no significant rise in reactor building pressure is expected on a loss of the RBS pumps. If the reactor building pressure is already so low that the RBS pumps lose NPSH, then the driving force for a release from the reactor building is limited.

The potential for losing the RBS pumps as a result of loss of NPSH during the sump recirculation phase has essentially no impact on the probability and consequences of risk-significant accidents analyzed in the PRA. The core damage frequency is unaffected by the situation and the public health risk could only be impacted in an insignificant manner.

Conclusion

Based on the above, Duke concludes that the proposed changes are justified and will have no impact on public health and safety.

FIGURE 1

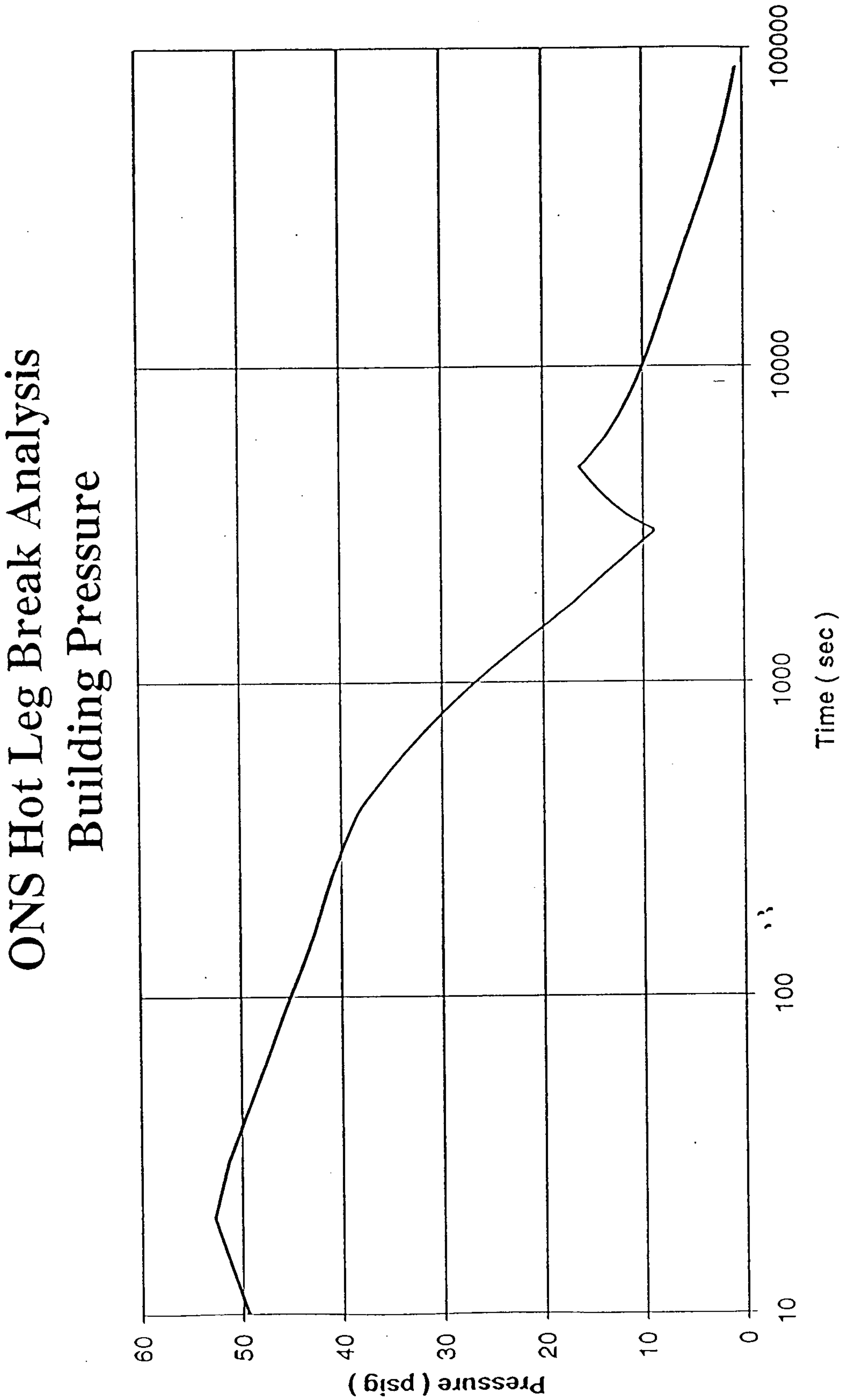
ONS Hot Leg Break Analysis
Building Pressure
Pressure (psig)
0
10
20
30
40
50
60
Time (sec)
10
100
1000
10000
100000

FIGURE 2

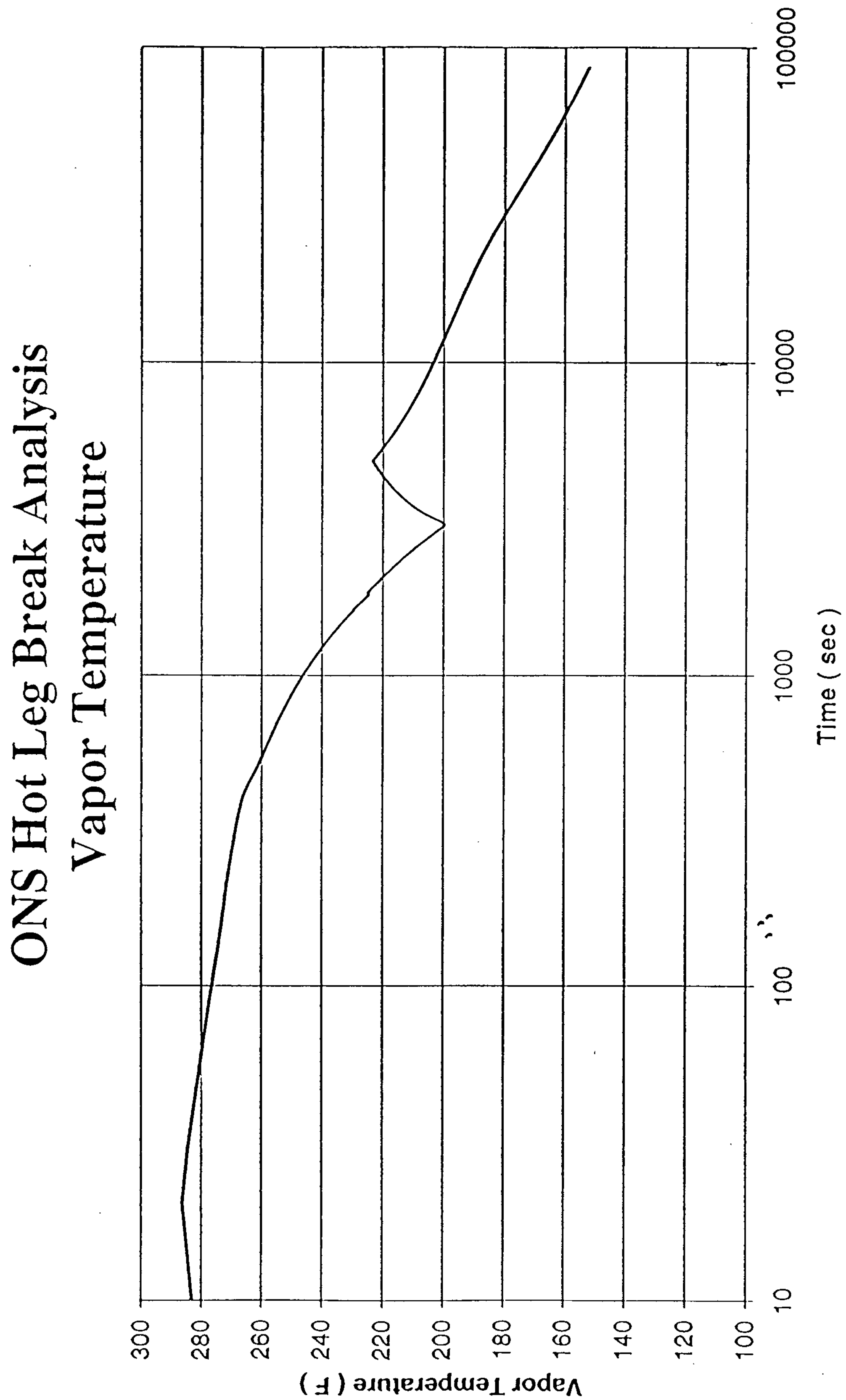
ONS Hot Leg Break Analysis
Vapor Temperature
Vapor Temperature (F)
300
280
260
240
220
200
180
160
140
120
100
Time (sec)
10
100
1000
10000
100000

FIGURE 3

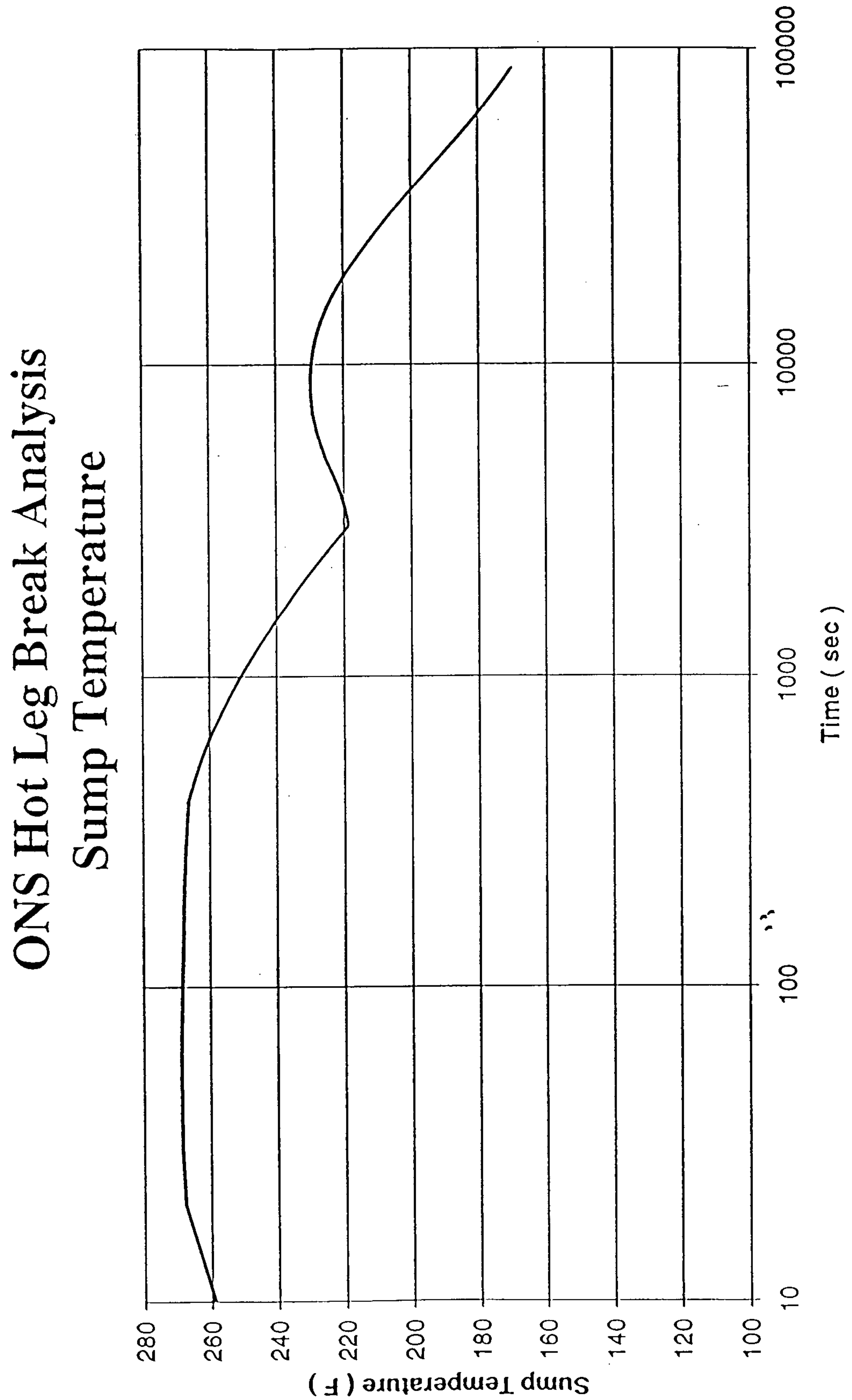
ONS Hot Leg Break Analysis
Sump Temperature
Sump Temperature (F)
280
260
240
220
200
180
160
140
120
100
Time (sec)
10
100
1000
10000
100000

FIGURE 4

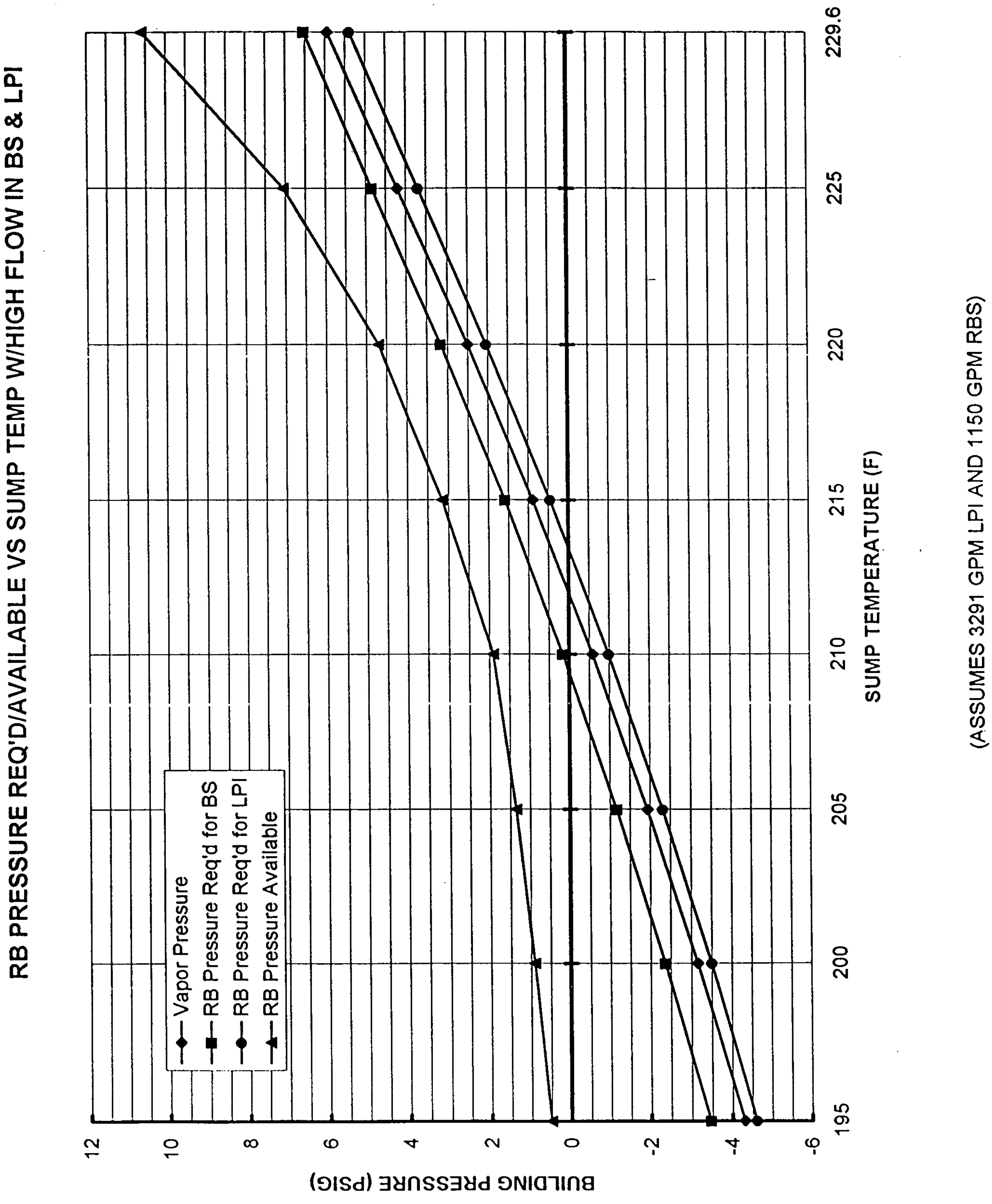
RB PRESSURE REQ'D/AVAILABLE VS SUMP TEMP W/HIGH FLOW IN BS & LPI
Vapor Pressure
RB Pressure Req'd for BS
RB Pressure Req'd for LPI
RB Pressure Available
BUILDING PRESSURE (PSIG)
12
10
8
6
4
2
0
-2
-4
-6
SUMP TEMPERATURE (F)
195
200
205
210
215
220
225
229.6
(ASSUMES 3291 GPM LPI AND 1150 GPM RBS)

TABLE 1

NPSH REQUIREMENTS FOR RBS & LPI OPERATION AT ELEVATED SUMP TEMPERATURES

SUMP TEMP (F)	NPSHr RBS (FT)	NPSHr Corrected RBS (FT)	NPSH AVAILABLE RBS (FT)	NPSHr LPI (FT)	NPSHr Corrected LPI (FT)	NPSH AVAILABLE LPI (FT)	NPSH REDUCTION FACTOR (FT)
195	17	16.50	25.98	13	12.50	24.70	0.50
200	17	16.40	24.21	13	12.40	22.93	0.60
205	17	16.30	22.35	13	12.30	21.08	0.70
210	17	16.25	20.46	13	12.25	19.20	0.75
215	17	16.15	19.93	13	12.15	18.67	0.85
220	17	16.10	19.83	13	12.10	18.57	0.90
225	17	16.00	21.35	13	12.00	20.09	1.00
229.6	17	15.90	25.80	13	11.90	24.54	1.10

BUILDING PRESSURE REQUIREMENTS FOR RBS & LPI OPERATION AT ELEVATED SUMP TEMPERATURES

SUMP TEMP (F)	BUILDING PRESSURE REQUIRED RBS (PSIG)	AVAILABLE RBS (PSIG)	REQUIRED LPI (PSIG)	AVAILABLE LPI (PSIG)	Psat (Gauge)
195	-3.46	0.50	-4.61	0.50	-4.31
200	-2.37	0.90	-3.51	0.90	-3.17
205	-1.17	1.35	-2.31	1.35	-1.93
210	0.15	1.90	-0.98	1.90	-0.58
215	1.58	3.15	0.45	3.15	0.89
220	3.16	4.70	2.03	4.70	2.49
225	4.84	7.05	3.71	7.05	4.21
229.6	6.52	10.60	5.40	10.60	5.94

OVERPRESSURE REQUIREMENTS FOR RBS & LPI OPERATION AT ELEVATED SUMP TEMPERATURES

SUMP TEMP (F)	OVERPRESS REQUIRED RBS (PSI)	OVERPRESS AVAILABLE RBS (PSI)	OVERPRESS REQUIRED LPI (PSI)	OVERPRESS AVAILABLE LPI (PSI)	Psat (Gauge)
195	0.00	4.81	0.00	4.81	-4.31
200	0.00	4.07	0.00	4.07	-3.17
205	0.00	3.28	0.00	3.28	-1.93
210	0.15	2.48	0.00	2.48	-0.58
215	0.69	2.26	0.00	2.26	0.89
220	0.67	2.21	0.00	2.21	2.49
225	0.63	2.84	0.00	2.84	4.21
229.6	0.58	4.66	0.00	4.66	5.94

NOTE: Required overpressure is required building pressure less Psat, except where Psat is negative. Where Psat is negative, required overpressure is required building pressure above atmospheric. Available overpressure is available building pressure less Psat.

Attachment 3

No Significant Hazards Consideration Evaluation

This proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to involve no significant hazards, in that operation of the facility in accordance with the license amendment request would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated?

   The reactor building spray (RBS) system is not considered as an initiator of any analyzed event, therefore, this change has no impact on the probability of an event previously analyzed.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the set points at which these actions are initiated. The proposed change permits limited reactor building overpressure to be credited in the calculation of available net positive suction head (NPSH) for the RBS pumps for a limited period of time during the sump recirculation phase. It is supported by calculations which demonstrate that adequate reactor building overpressure will be available to ensure the RBS system will be capable of performing its safety function. Thus, the proposed change does not significantly increase the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from the accidents previously evaluated?

   The proposed change permits limited reactor building overpressure to be credited in the calculation of available NPSH for the RBS pumps for a limited period of time during the sump recirculation phase. It does not involve a physical alteration of the plant. The proposed change is supported by calculations which demonstrate that adequate reactor building overpressure will be available to ensure the RBS system will be capable of performing its safety function. This change will not

alter the manner in which the RBS system is initiated, nor will the function demands on the RBS system be changed. Thus, the proposed change does not create the possibility of a new or different kind of accident.

3. Involve a significant reduction in a margin of safety?

The proposed change permits limited reactor building overpressure to be credited in the calculation of available NPSH for the RBS pumps for a limited period of time during the sump recirculation phase. Crediting a slight amount of overpressure does not result in a significant reduction in the margin of safety, because conservative analyses demonstrate that adequate reactor building overpressure will be available to ensure the RBS system will be capable of performing its safety function. Thus, the proposed change does not involve a significant reduction in a margin of safety.

Duke has concluded based on the above information that there are no significant hazards involved in this LAR.

Attachment 4

Environmental Assessment

Pursuant to 10 CFR 51.22 (b), an evaluation of the license amendment request (LAR) has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) 9 of the regulations. The LAR does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the determination of no significant hazards contained in Attachment 3.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   This LAR will not change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

   This LAR will not increase the individual or cumulative occupational radiation exposure.

In summary, this LAR meets the criteria set forth in 10 CFR 51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.