PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

APR 0 2 1996

LR-N96080

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

LER 272/95-016-01
SALEM GENERATING STATION - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

This Licensee Event Report Supplement entitled "Difference Between Containment Design Parameters and Accident Analysis" is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73(a)(2)(i)(B).

Sincerely,

Clay Warren
General Manager -
Salem Operations

Attachment

SORC Mtg. 96-038

DVH/tcp

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Attachment A

The following item represents commitments that Public Service Electric & Gas (PSE&G) made to the Nuclear Regulatory Commission (NRC) relative to this LER (272/95-016-01). The commitments are as follows:

1. Procedure NAP 59, "10CFR50.59 Reviews and Safety Evaluations" will be revised to address text search recommendations. This procedure revision will be completed by July 31, 1996.

2. UFSAR changes will be completed to address the containment design basis for the MSLB events prior to restart.

3. A License Change Request to address the peak containment temperature discrepancies between the design basis and that stated in the Technical Specifications will be implemented by entry into mode.

4. Modifications for the reactor coolant pump platforms for both units will be completed by the restart of each unit.

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)
DIFFERENCE BETWEEN CONTAINMENT DESIGN PARAMETERS AND ACCIDENT ANALYSIS

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									SALEM, UNIT 2	05000311
07	20	95	95	– 016 –	01	04	01	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)			20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
DENNIS V. HASSLER, LER COORDINATOR	609-339-1989

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On July 20, 1995, during a review of the containment liner design basis documents, a discrepancy was identified between the design basis for the containment structure as described in the Technical Specifications; the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis for a Main Steam Line Break Accident and Design Basis Loss of Coolant Accident; and the existing design calculations for the containment structure. The containment liner was determined to enter the yield range during a main steam line break accident. This is outside of the current design basis.

The root causes for this condition were the initial licensing basis was not consistent with respect to all design basis considerations, and inadequate 10CFR50.59 safety evaluations for changes in the containment temperature profiles did not consider all sections of the Technical Specifications and UFSAR affected by the changes. Corrective actions include UFSAR and Technical Specifications changes, procedure enhancements and modifications.

This is reportable under 10 CFR50.73(a)(2)(ii), a condition that was outside of the design basis of the plant.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT and SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes and component function identifier codes appear in the text as {SS/CC}.

CONDITIONS PRIOR TO OCCURRENCE

Both Units were in a self-imposed extended shutdown.

Mode: 5 Reactor Power: -0-% Unit Load: -0- MWe

DESCRIPTION OF OCCURRENCE

On July 20, 1995, during a review of the containment liner {NH/-} design basis documents, a discrepancy was identified between the design basis for the containment structure as described in the Technical Specifications; the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis for a Main Steam Line Break Accident (MSLB) and Design Basis Loss of Coolant Accident (LOCA); and the existing design calculations for the containment structure. Technical Specifications Section 5.2.2 describes the containment structure as being designed for a maximum pressure of 47 psig and an air temperature of 271 degrees Fahrenheit (F). The transient following a SLBA produces temperatures in containment which peak at 351.3 degrees F. The UFSAR also states that the maximum temperature at the uninsulated section of the liner under accident conditions is 246 degrees F. Due to the lack of documentation that evaluated and accepted these apparent discrepancies, a four hour report was made in accordance with 10 CFR 50.72.b.2.i.

ANALYSIS OF OCCURRENCE

The UFSAR Chapter 15 accidents reviewed to define the licensing basis peak containment temperature profile were the LOCA and the MSLB. Section 15.4.8.2.4 of the UFSAR discusses the results of the spectrum of steam line breaks that was analyzed for peak conditions. The peak containment temperature predicted by these analyses is listed as 351.3 degrees Fahrenheit. This compares with the Technical Specifications value of 271 degrees Fahrenheit. A License Change Request to address the peak containment temperature discrepancies between the design basis and that stated in the Technical Specifications will be submitted.

Revision 0 of this LER stated that the effects of the elevated temperature on equipment evaluated under the EQ program for the steam line break were found to be acceptable. Additionally, Rev 0 stated that results of the steam line break analysis impact on the containment liner {NH/-LNR}, containment spray piping supports{BE/SPT}, containment hatches {NH/D}, floor framing/miscellaneous steel and other steel structures would be reported in the supplement.

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Containment and Liner -

The analysis has determined that a MSLB will cause stresses in the liner plate to enter the yield range. The liner entering the yield range is outside of the current design basis. Although the liner plate may yield, the containment will still be able to perform its function because the pressures and temperatures do not overstress the reinforced concrete shell and the steel liner is not needed for strength. The analysis concluded that the liner will maintain leak tight integrity. UFSAR changes will be completed to address the containment design basis for MSLB events.

Containment Spray Piping Supports -

A review of the steel structures in the containment was performed to address the accident thermal loads. This review determined that some connections on the Unit 1 containment spray support steel would cause local yielding of the liner and embedments where the spray pipe support framing is attached. A modification to these connections has been completed which precludes the liner from yielding. No concerns were identified for the Unit 2 containment spray support steel.

Containment Hatches -

The containment equipment hatch and personnel hatch {NH/D} have seals made of silicone rubber gaskets. An evaluation has concluded that the seals will meet design basis requirements.

Floor Framing/Miscellaneous Steel -
Other Steel Structures -

A review of the steel structures in the containment was performed to address the accident thermal loads. This review determined that floor framing, which utilizes slotted holes and pin connections, allow for thermal growth and therefore is adequate. However, the Unit 1 and 2 reactor coolant pump platforms could not be demonstrated to accommodate the thermal growth, and modifications to address this deficiency have been initiated.

CAUSE OF OCCURRENCE

The root causes for this condition were:

- The initial licensing basis was not consistent with respect to all design basis considerations.

- Inadequate 10CFR50.59 safety evaluations for changes in containment temperature profiles that did not consider all sections of the Technical Specifications and UFSAR. As a result, the safety evaluations were not reviewed by all affected engineering disciplines.

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PRIOR SIMILAR OCCURRENCE

During the previous two years there were three LERs that attributed their root cause to design deficiencies. However, the event in this LER involved not only design issues but also the design basis change process and therefore is not considered similar to the other three LER events.

SAFETY CONSEQUENCES AND IMPLICATIONS

There were no consequences with the design discrepancy described in this LER because a MSLB or LOCA have not occurred. As previously described, the containment would have performed its intended function. For the containment spray piping supports, local yielding of the liner would have occurred, however the configuration of the framing is such that there would not have been any catastrophic failure of the containment spray piping supports. However, during a MSLB and LOCA, the reactor coolant pump platforms structure may have exceeded yield and ultimate values. If this had occurred, other systems may have been damaged. The failure mechanism of the platform is not known and therefore the extent of damage to other systems is indeterminate.

CORRECTIVE ACTIONS

1. A Chapter 15 licensing basis accident analysis comprehensive review was performed to determine if other similar design and licensing basis discrepancies existed. No similar design and licensing basis discrepancies were discovered.

2. A lessons learned document was sent to preparers and peer reviewers of 10CFR50.59 safety evaluations addressing the need to do a text search of the Technical Specifications and UFSAR that may be affected by a proposed change. Procedure NAP 59, "10CFR50.59 Reviews and Safety Evaluations" will be revised to address the text search recommendations. This procedure revision will be completed by July 31, 1996.

3. UFSAR changes will be completed to address the containment design basis for the MSLB events prior to restart.

4. A License Change Request to address the peak containment temperature discrepancies between the design basis and that stated in the Technical Specifications will be implemented by entry into mode 4.

5. Modifications for the Unit 1 containment spray piping supports have been completed. Modifications for the reactor coolant pump platforms for both units will be completed by the restart of each unit.