NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Hope Creek Generating Station	05000354	1 OF 6

TITLE (4)
CORE SPRAY NOZZLE WELD THROUGH-WALL LEAK

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
										05000
09	19	97	97	-- 023 --	01	11	28	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
POWER LEVEL (10)	0		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)
Charlie Manges, Station Licensing Engineer - Hope Creek	(609) 339-3234

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
B	BM	PSP	G080	N					

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

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

At 1305 on 9/19/97, through-wall leaks were identified on the Core Spray (CS) nozzle-to-safe-end weld associated with the "A" CS subsystem. The condition placed the operability of the "A" CS subsystem in question, and the plant entered Limiting Condition for Operation (LCO) 3.5.2, "ECCS - Shutdown" and LCO 3.4.8, "Structural Integrity." The cause of the through-wall leakage has been attributed to intergranular stress corrosion cracking (IGSCC) in the Alloy 182 weld material. The weld was a field weld performed during the replacement of safe-ends in the early 1980s. The cause of the failure to detect the IGSCC during the last UT examination in 1995 was human error with a contributing factor being lack of use of information on weld configuration and history during review of the data. Corrective actions taken include nondestructive examination of the subject weld, review of the 1995 ultrasonic test (UT) data for the weld as well as for other Category "D" welds (i.e., welds that are not made with resistant materials and that have not been given stress improvement treatment but that have been inspected and found to be free of cracks), review of radiographs for field welds made for the safe end replacements in the early 1980s, UT examination of additional Category D welds, and repair of the leak using an engineered weld overlay. Corrective actions planned include establishing an as-built configuration for each weld and making that information available to personnel analyzing UT data and establishing requirements to ensure independent evaluation of UT data for Category D welds.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)
Core Spray System - EIIS Identifier (BM/PSP)*

* Energy Industry Identification System (EIIS) codes and component function identifier codes appear as (SS/CC).

IDENTIFICATION OF OCCURRENCE

Event Date: September 19, 1997
Problem Report: 970919211

CONDITIONS PRIOR TO OCCURRENCE

The plant was in OPERATIONAL CONDITION 5 (REFUELING) for Hope Creek's seventh refueling outage (RFO7). There were no structures, systems, or components that were inoperable at the beginning of the event that contributed to the event.

DESCRIPTION OF OCCURRENCE

On September 19, 1997, during a routine tour of the Hope Creek drywell, personnel noticed dripping water and observed that the water was coming from the N5BSE core spray nozzle safe-end weld associated with the "A" Core Spray (CS) subsystem (BM/PSP). Radiochemical analyses performed on samples of water taken from the drywell indicated that the fluid was reactor coolant. Further investigation revealed the presence of through-wall leaks at the top of the core spray nozzle-to-safe-end weld. This placed the operability of the "A" CS subsystem in question, and the unit entered Limiting Condition for Operation (LCO) 3.5.2, "ECCS - Shutdown" and LCO 3.4.8, "Structural Integrity." The unit was in Operational Condition 5 at the time of discovery. Subsequently, the N5BSE weld was nondestructively examined using a 45 degree shear wave and 45 and 60 degree refracted longitudinal wave search units. The data was evaluated by General Electric (NDE services supplier), PSE&G and EPRI NDE Center personnel. An ultrasonic reflector with indications exhibiting characteristics of IGSCC was identified.

DESCRIPTION OF OCCURRENCE (Continued)

The subject weld was a field weld performed during the replacement of the furnace-sensitized safe-ends in the early 1980s. No indications associated with Intergranular Stress Corrosion Cracking (IGSCC) had been recorded as a result of the last nondestructive examination (ultrasonic test) of the weld performed during the sixth refueling outage (RFO6) in 1995. During the investigation, PSE&G and EPRI NDE Center personnel reviewed the RFO6 ultrasonic test (UT) data for the subject weld and identified an indication of through-wall penetration to a depth of approximately 0.56 inches (nominal weld thickness is 1.25 inches).

On September 19, 1997, at 1515, a four hour notification was made to the NRC in accordance with 10CFR50.72(b)(2)(i). This event is being reported pursuant to 10CFR50.73(a)(2)(ii) as a condition that resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded. Specifically, the through-wall leak is considered a significant degradation of the reactor coolant system pressure boundary. This event is also being reported pursuant to 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications (TSs). Based on review of the 1995 UT data for the subject weld, there is reason to believe that, TS 3.4.8, "Structural Integrity", was violated.

CAUSE OF OCCURRENCE

The cause of the through-wall leakage at the N5BSE core spray nozzle-to-safe-end weld has been attributed to IGSCC in the Alloy 182 weld material. Additional details concerning the root cause analysis and the basis for PSE&G's root cause conclusions are provided in PSE&G Letter LR-N970667 dated October 9, 1997.

The cause of the failure to detect IGSCC during the last UT examination in 1995 was human error in that the certified Level 2 NDE analyzer incorrectly evaluated the ultrasonic indications. Contributing factors to the human error included the lack of use of information on weld configuration and history. The evaluation process was also determined to be vulnerable to a single failure.

PREVIOUS OCCURRENCES

A review of LERs issued in the last two years for Hope Creek did not identify any similar occurrences.

ASSESSMENT OF SAFETY CONSEQUENCES

There were no actual consequences and no impact on public health and safety. The condition did not result in a pipe break, there was no radioactive release, and the leakage was within allowable Technical Specification limits.

Further propagation of the crack could have resulted in an increase in the unidentified leakage in the drywell. The increase in the unidentified leakage would have required a reactor shutdown and cooldown before a catastrophic failure would have occurred.

A more rapid propagation of the crack could have occurred due to an increase in loading caused by a transient such as Main Steam Isolation Valve (MSIV) closure or a seismic event. Such a propagation could have resulted in additional leakage. The possible extent of propagation of the crack is bounded by a CS line break as detailed below.

CS Line Break: A CS spray line failure would constitute a loss of coolant accident (LOCA). A failure of the CS line in the drywell would not result in fuel damage. The calculated peak cladding temperature (PCT) for the CS line failure event, assuming a single failure, is less than 1500°F. Following the break, the reactor would trip and the containment would automatically isolate. The offsite dose for such a break is bounded by the UFSAR dose analyses.

ASSESSMENT OF SAFETY CONSEQUENCES (Continued)

CS Line Break Concurrent With Design Basis LOCA: This CS line break scenario is also evaluated with the conservative licensing basis assumptions discussed in Section 6.3 of the UFSAR. The design basis LOCA, with the assumed limiting single failure, is mitigated by three LPCI trains and one CS subsystem (HPCI, one CS subsystem and one LPCI train fail due to the assumed single failure). With the additional failure at the CS nozzle, however, the CS train would also become unavailable for core cooling. Therefore, only three LPCI trains would be available for mitigation. Although some RPV inventory would be lost through the CS break, the higher elevation of the break would aid in the RPV depressurization, permitting an earlier initiation of injection from the LPCI. Since the CS train would not be available for injection, the reflooding of the RPV would be slower. The hot spot would remain uncovered for a longer time until LPCI injection alone would reflood the vessel. This would result in a higher PCT; however, the resulting PCT would remain below the licensing limit of 2200°F, and the offsite dose would be bounded by the UFSAR dose analysis.

A best estimate LOCA analysis has been performed by General Electric (GE) using the NRC approved SAFER/GESTR methods. Although these Hope Creek specific analyses have not been submitted to the NRC for approval, these analyses show relative insensitivity of injection flowrate to the PCT. This insensitivity is mainly due to the high capacity of the Hope Creek Emergency Core Cooling System (ECCS). With SAFER/GESTR methods, for limiting UFSAR scenarios described above the licensing basis, the PCT remains below 1600°F.

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

CORRECTIVE ACTIONS

1. The UT data from the RF06 examination of the remaining 19 Category "D" welds was reviewed by GE and EPRI NDE Center personnel looking for conditions similar to those observed in the N5BSE weld. The reviews resulted in identification of one weld (N2J) with sufficient indication to warrant further investigation. This was one of the welds examined under Corrective Action 3, and as indicated below, further examination exhibited no indications associated with IGSCC.

2. The construction radiographs of field welds made for safe-end replacements in the early 1980s were reviewed by PSE&G. The radiograph of the N5BSE weld showed indications of lack of fusion above the root. Other welds were acceptable.

3. UT examinations of additional welds were completed and exhibited no indications associated with IGSCC.

4. The through-wall leak in the N5BSE weld was repaired using an engineered weld overlay.

5. Requirements will be established to ensure independent evaluation of UT data for Category D welds prior to conducting UT examinations during the eighth refueling outage.

6. PSE&G will establish an as-built configuration for each weld and make the information readily available to personnel analyzing UT data. This action will be completed prior to conducting UT examinations during the eighth refueling outage.