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 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

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)
Pressurizer Level Controlled Higher than Assumed in 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
05	20	97	97	-- 025 --	00	06	19	97	Unit 2	05000301
									FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9): N

POWER LEVEL (10): 000

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)
	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)
Curtis A. Castell	(414) 221-2019

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 May 20, 1997, while Point Beach Nuclear Plant (PBNP) Unit 1 was in a cold shutdown condition and Unit 2 was in a defueled condition, it was discovered that three Operating Procedures (OP) allow manual control of pressurizer level 10% higher than assumed in the PBNP FSAR section 14.2.5 rupture of steam pipe analysis assumption of 20%. This condition was discovered by Reactor Engineering personnel while reviewing operations and reactor engineering procedures in anticipation of the impending restart of Unit 2. These procedures were changed in the mid-1980's to allow manual control of pressurizer level at 30% in lieu of the automatic program level of 20% at zero power operation. This condition was caused by inappropriately changing the procedure without adequate consideration of potential affects on the PBNP accident analyses. The affected procedures will be revised prior to restart of Unit 2. It was determined that the consequences of the PBNP FSAR section 14.2.5 "Rupture of a Steam Pipe" analysis would not exceed 10 CFR 100 limits, even if some fuel damage could occur based on the use of a higher pressurizer level.

Event Description:

On May 20, 1997, while Point Beach Nuclear Plant (PBNP) Unit 1 was in a cold shutdown condition and Unit 2 was in a defueled condition, it was discovered that Operating Procedures OP-1A, "Cold Shutdown to Low Power Operation," OP-1C, "Low Power to Normal Power Operation," and OP-3A, "Normal Power to Low Power Operation" allow manual control of pressurizer level 10% higher than assumed in the PBNP FSAR section 14.2.5 rupture of steam pipe analysis assumption of 20%. This condition was discovered by Reactor Engineering personnel while reviewing operations and reactor engineering procedures in anticipation of the impending restart of Unit 2.

These procedures were changed in the mid-1980's to allow manual control of pressurizer level at 30% in lieu of the automatic program level of 20% at zero power operation. This change was implemented to provide more operational margin between the chemical and volume control system letdown isolation level setpoint at 12% and the actual level being maintained in the pressurizer.

Cause:

This condition was caused by inappropriately changing the procedure without adequate consideration of potential affects on the PBNP accident analyses.

Corrective Actions:

The affected procedures, OP-1A, "Cold Shutdown to Low Power Operation," OP-1C, "Low Power to Normal Power Operation," and OP-3A, "Normal Power to Low Power Operation," will be revised to discontinue the practice of manually controlling pressurizer level at 30% prior to restart of Unit 2.

Recent improvements to the procedure change and 10 CFR 50.59 review processes used at PBNP are expected to minimize the possibility of this condition recurring.

A root cause evaluation is being completed. Additional corrective actions will be taken, as appropriate, from recommendations contained in the root cause evaluation.

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TEXT CONTINUATION

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Reportability:

This Licensee Event Report is being submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(ii)(A), "Any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly compromised plant safety."

Component and System Description:

Pressurizer level is automatically controlled using a linear program from 20% at the zero power average temperature condition of 547°F to 45.8% at the full power average temperature condition of 570°F. Below 547°F level is limited to 20% and above 570°F level is limited to 45.8%. Level deviations from the program setpoint causes increasing or decreasing charging pump flow based on actual pressurizer level below or above the program level.

Safety Assessment:

The PBNP FSAR section 14.2.5 "Rupture of a Steam Pipe" analysis assumes a 30% pressurizer level, which is based on 20% program level plus 10% allowance for conservatism. By changing the program level to 30% and continuing to assume the 10% allowance, the pressurizer level could be assumed to be 40%. If 40% was used in the analysis, the consequences of the steam line rupture may increase. In particular, higher pressurizer level could delay depressurization which reduces the safety injection flow and delays the onset of accumulator injection. Analyses to quantify the effect of the higher pressurizer level have not been performed.

It was determined that the consequences of the PBNP FSAR section 14.2.5, "Rupture of a Steam Pipe" analysis would not exceed 10 CFR 100 limits, even if some fuel damage could occur based on the use of a higher pressurizer level. This is based on the judgment that the consequences of the large break loss of coolant accident (PBNP FSAR section 14.3.5), which is based on the release of all volatile fission products into the containment and subsequent release of 0.4 weight%/day for the first 24 hours and 0.2 weight%/day for the next 29 days, would be greater. The results of the large break loss of coolant analysis show compliance with 10 CFR 100. Therefore, it is not expected that operation with increased

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pressurizer level during the shutdown condition would cause consequences of the PBNP FSAR section 14.2.5 "Rupture of a Steam Pipe" analysis to exceed 10 CFR 100 limits, even if some fuel damage could occur based on the higher pressurizer level.

System and Component Identifiers

The Energy Industry Identification System component function identifier for each component/system referred to in this report are as follows:

Component/System	Identifier
Pressurizer	PZR
Charging pump	P
Level Controller	LC
Main Steam System	SB
Reactor Coolant System	AB

Similar Occurrences:

A search was conducted of previously submitted licensee event reports similar to this situation for PBNP. The specific criterion used was based on a search for licensee event reports that were submitted due to plant procedures that allowed or caused the plant to be not in accordance with accident analysis assumptions.

LER 266/301-84-005-00 identified a condition that allowed operation of the units such that the conditions of FSAR section 14.1.1, "Uncontrolled RCCA Withdrawal from a Subcritical Condition," analysis could be invalidated.