Wisconsin Electric
POWER COMPANY

Point Beach Nuclear Plant
6610 Nuclear Rd., Two Rivers, WI 54241

(414) 755-2321

PBL 97-0036

February 4, 1997

Document Control Desk
US NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Ladies/Gentlemen:

DOCKET 50-266 AND 50-301
LICENSEE EVENT REPORT 97-001-00
SAFETY INJECTION DELAY TIMES
EXCEED DESIGN BASIS VALUES
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is Licensee Event Report 97-001-00 for Point Peach Nuclear Plant, Units 1 and 2. This report is provided in accordance with 10 CFR 50.73(a)(2)(ii)(B), "a condition that was outside the design basis of the plant." This report describes a condition where delay times assumed for the high and low head safety injection flow in the Large Break Loss of Coolant Accident analysis were not conservative.

If you require additional information, please contact us.

Sincerely,

Thomas G. Malanowski
for Douglas F. Johnson
Manager-Regulatory Services & Licensing

JAK

Enclosure

cc: NRC Resident Inspector
NRC Regional Administrator, Region III

IE22

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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Point Beach Nuclear Plant, Unit 1	05000266	1 OF 5

TITLE (4)

Safety Injection Delay Times Exceed Design Basis Values

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
									PBNP Unit 2	05000301
01	08	97	97	-- 001 --	00	02	04	97	FACILITY NAME	DOCKET NUMBER 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	N							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)	90		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
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)	or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeff Kos, Design Basis Engineer	(414) 221-4917

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 January 8, 1997, with Unit 1 operating at 90% power and Unit 2 in a refueling shutdown, licensee engineers determined that the delay times assumed for high and low head safety injection (SI) flow in the Large Break Loss of Coolant Accident (LBLOCA) analysis were not conservative. The LBLOCA licensing basis analysis assumed that the high and low head SI systems were capable of providing full flow within five and ten seconds respectively. A conservative licensee evaluation concluded that the total delay times may be as high as 8.0 seconds for high head SI and 23.7 seconds for low head SI. The delay time assumptions for the licensing basis analysis did not account for time delays associated with SI signal processing, sequencer delay time uncertainty, or an increased time for pump acceleration to full speed due to degraded voltage conditions. A Westinghouse safety assessment concludes that the increased safety injection delay times do not result in exceeding any design or regulatory limit for Point Beach Units 1 and 2.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Point Beach Nuclear Plant, Unit 1	05000266	97	- 001 -	00	2 OF 5

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

Event Description:

On January 8, 1997, with Unit 1 operating at 90% power and Unit 2 in a refueling shutdown, licensee engineers determined that the delay times assumed for the high and low head safety injection (SI) flow in the existing LBLOCA analysis were not conservative. This condition was discovered during a review of the stroke time performance requirements for the SI system valves SI-852A,B

The existing LBLOCA analysis assumes that the high and low head SI systems are capable of providing full flow within five and ten seconds respectively. These assumptions are based on a 5 second delay for the high head SI pump to come up to speed and a 10 second delay for the low head SI pumps to load on the sequencer and come up to speed. This delay time represents the time from when the SI setpoint is reached to the time when the pumps are capable of providing full flow. The delay time assumptions for the existing licensing basis analysis do not account for (1) time delays associated with SI signal processing, (2) sequencer delay time uncertainty, or (3) an increased time for pump acceleration to full speed due to degraded voltage conditions. When these delays were combined, the licensee evaluation concluded that the total delay times may be as high as 8.0 seconds for high head SI and 23.7 seconds for low head SI. The impact of the increased delay times on the LBLOCA analysis is described below.

The applicable acceptance criteria for the LBLOCA analysis is a peak cladding temperature (PCT) of 2200°F, as identified in 10 CFR 50.46. The most recent submittal (which does not include the results of the additional time delay) to the NRC on ECCS Evaluation Model changes reflects a 109°F penalty in margin allocations to the licensing basis analysis LBLOCA PCT of 2028°F, resulting in a PCT of 2137°F. Westinghouse formally evaluated the increase in the safety injection delay times on the LBLOCA analysis. The revised cumulative LBLOCA PCT is 2181°F, based on a 44°F penalty due to the longer SI delay times. This evaluation concludes that the calculated maximum fuel element cladding temperature does not exceed 2200°F, the calculated total local oxidation of the cladding nowhere exceeds 0.17 times the total cladding thickness before oxidation, and the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Conformance to the criteria above demonstrates that the core geometry is maintained such that the core remains amenable to cooling. The increase in SI system delay will not impact the long-term ability to maintain core temperature at an acceptably low value and to remove decay heat for an extended period of time required by the long-lived radioactivity remaining in the core. Therefore, the specific safety limits defined by the ECCS Acceptance Criteria of 10 CFR 50.46 for the LBLOCA licensing basis analysis including penalties are met.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 1	05000266	97	- 001 -	00	3 OF 5

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

The IEEE Standard 803A-1983 component identifier for this report is:

Pump (p)

Component and System Description:

The SI system shall deliver borated cooling water to the reactor coolant system during the injection phase of SI to support core cooling and to ensure adequate shutdown margin in the event of a main steam line break. The licensing basis accident analyses, as described in Chapter 14 of the FSAR, ensure that the SI system is capable of performing these safety-related functions. The LBLOCA analysis (FSAR Section 14.3.2), SBLOCA (FSAR Section 14.3.1), MSLB (FSAR Section 14.2.5), SGTR (FSAR Section 14.2.4) and the Containment Integrity Evaluation (FSAR Section 14.3.4) assume SI actuation resulting in safety injection flow to the reactor coolant system.

PBNP Technical Specification 15.4.6.A.2 requires that a test be performed to demonstrate the ability of a diesel generator to automatically start, shed load, and restore particular vital equipment to operation following an actual interruption of normal AC station service power supply to associated engineered safety systems busses together with a simulated safety injection signal. The Technical Specifications require that the test be conducted to assure that the diesel generator will start and assume required load in accordance with the timing sequence listed in FSAR Section 8.2 after the initial starting signal. These acceptance criteria are also listed in Operations Refueling Test (ORT) 3, Appendix C, Attachment 1. The acceptance criteria from this test are used in evaluating the affects on the LBLOCA analysis.

Cause:

The existing LBLOCA analysis provides inadequate margin in the SI system delay time assumption to accommodate for time delays associated with SI signal processing, sequencer delay time uncertainty, and an increased time for pump acceleration to full speed due to degraded voltage conditions. The basis for the delay times are a result of typical vendor methodology which may result in a non-conservative LBLOCA analysis with respect to SI system actuation delay times. Also, the acceptance criteria for the SI sequence test of ORT 3 as described in FSAR section 8.2 and ORT 3 Appendix C Attachment 1 should have been thoroughly reviewed to assure that the times remain within the LBLOCA analysis assumptions.

Safety Assessment:

The specific safety limits defined by the ECCS Acceptance Criteria of 10 CFR 50.46 are met. Therefore, the health and safety of the public is not compromised by the conditions described herein.

Corrective Actions:

The following corrective actions have been taken or are planned to address this event:

1. Westinghouse has formally evaluated the increase in the safety injection delay times on the LBLOCA analysis. This evaluation determined that the specific safety limits defined by the ECCS Acceptance Criteria of 10 CFR 50.46 impacted by an increase in safety injection delay time for the LBLOCA analysis are met.

2. Licensee engineers will prepare FSAR change requests to reflect this LBLOCA evaluation.

3. Licensee engineers will review the SBLOCA (FSAR Section 14.3.1), MSLB (FSAR Section 14.2.5), SGTR (FSAR Section 14.2.4) and the Containment Integrity Evaluation (FSAR Section 14.3.4) accident analyses. These analyses assume SI actuation resulting in safety injection flow to the reactor coolant system. This review will verify the assumptions associated with SI system delay times are appropriately conservative.

4. The acceptance criteria for SI sequence test of ORT 3 as described in FSAR section 8.2 and ORT 3, Appendix C, Attachment 1, will be reviewed to assure that the delay times associated with sequenced safeguards components remain within the FSAR Chapter 14 accident analysis assumptions.

Reportability:

A 1-hour prompt notification per 10 CFR 50.72(a)(2)(ii)(B) was reported to the NRC duty officer at 1344 CST on January 8, 1997. This Licensee Event Report is being submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(ii)(B), "A condition that was outside the design basis of the plant."

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 1	05000266	97	- 001 -	00	5 OF 5

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

Similar Occurrences:

A search of the LER database discovered a prior occurrence of non-conservative accident analysis assumptions resulting in a system being declared in a condition that is outside the design basis.

LER	Title
266/96-015-00	Main Steam Safety Valve Lift Setpoints Exceed Design Basis Values