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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2001-044

July 18, 2001

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, D.C. 20555

10 CFR 50.73

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
LICENSEE EVENT REPORT 266/2001-003-00
CONTAINMENT RESPONSE FOR MSLB MAY EXCEED
DESIGN PRESSURE OF 60 PSIG
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is Licensee Event Report 266/2001-003-00 for the Point Beach Nuclear Plant (PBNP), Units 1 and 2. The subject condition was determined to be reportable under 10 CFR 50.73(a)(2)(ii)(B) as: "Any event of condition that resulted in: (B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety." This LER documents our preliminary evaluation that in the event of a main steam line break accident (MSLB) with a coincident failure of a main feedwater regulating valve to close, the internal pressure of the containment structure may briefly exceed the FSAR design pressure of 60 psig. Although our evaluation of this condition indicates that the integrity of the containment structure would not be challenged by this postulated event, we are conservatively providing this event report as a follow-on to our June 7, 2001, 10 CFR 50.72 notification.

New corrective action commitments have been identified with italics in the Corrective Actions section of this report.

Please contact us if you require additional information concerning this report.

Sincerely,

A. J. Cayia
Plant Manager

Enclosure

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW
INPO Support Services

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(1-2001)

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 4

TITLE (4)

CONTAINMENT RESPONSE FOR MSLB MAY EXCEED DESIGN PRESSURE OF 60 PSIG

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Point Beach Unit 2	05000301
06	07	2001	2001	003	00	07	20	2001	FACILITY NAME	DOCKET NUMBER 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : (Check all that apply) (11)

OPERATING MODE (9)	1		20.2201(b)		20.2203(a)(3)(ii)	X	50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
POWER LEVEL (10)	100		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)	50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)	73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)	73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)	
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)	
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)	
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)	
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Charles Wm. Krause	(920) 755-6809

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		10	30	2001

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

This report documents our preliminary evaluation that in the event of a main steam line break accident (MSLB) with a coincident failure of a main feedwater regulating valve (MFRV) to close, the internal pressure of the containment structure may briefly exceed the FSAR design pressure of 60 psig. The MSLB with MFRV failure to close has not been previously evaluated for PBNP at the presently licensed thermal power. However, an analysis of this accident under up-rated licensed power conditions has been completed. Based on an evaluation of the information provided in that analysis, the plant may be in an unanalyzed condition. Our evaluation of this condition indicates that the integrity of the containment structure would not be challenged by this postulated event, and; therefore, the safety significance of this condition is low. We are conservatively providing this event report as a follow-on to our June 7, 2001, 10 CFR 50.72 notification. We are planning to perform a revised analysis of the MSLB with MFRV failure to close under the currently licensed conditions.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Point Beach Nuclear Plant Unit 1	05000266	2001	- 003 -	00	2 OF 4

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

Event Description:

During a review of the containment analysis conducted to support a possible reactor power up-rate of the Point Beach Nuclear Plant (PBNP), a potential non-conservatism was identified. The analysis of concern is a main steam [SB] line break (MSLB) inside containment with an assumed single failure of a main feedwater regulating valve [FCV] to close. The existing analysis for the MSLB accident, which is based on a generic Westinghouse two loop analysis, does not assume the failure of a main feedwater regulating valve (MFRV) to close and thus does not fully account for the volume of high temperature feedwater [SJ] that could flash to steam and be released inside the containment [NH] during the MSLB accident. This was determined to be a potential unanalyzed condition which may have safety significance and was reported to the NRC in accordance with 10 CFR 50.72(b)(3)(ii)(B) on June 7, 2001 (Event Notification 38057).

In 1999, the results of a containment pressure response study (WCAP 15153,"Point Beach Unit 1 and 2 Steamline Break and Containment Integrity Analysis") to support a proposed power up-rate project was originally reviewed. This study showed for the worst case MSLB accident conditions that the containment pressure would peak at 64.2 psig, exceeding the containment design pressure of 60 psig. This analysis had been performed with an assumed uprated power of 1650 MWth. PBNP is currently licensed to a maximum thermal power limit of 1518.5 MWth . At the time of the 1999 review, the increased containment pressure for the MSLB was attributed to the higher thermal power rating assumed for the up-rate project. This condition was documented in the Point Beach corrective action program (CR 99-0153) with resolution of the issues deferred to the continuing evaluations of the power up-rate project.

In April 2000, this issue was revisited when discussions with another licensee brought into question the licensing basis for the PBNP MSLB. This review was also documented in the PBNP corrective action program (CR 00-1304). The review of the WCAP at that time identified that for the up-rated capacity conditions, the increase in containment peak pressure from the worse case containment safeguards failure (Loss of one train of containment spray and two containment fan coolers) to the single failure of a MFRV to close is only 6.7 psig. Adding this increase to the previously evaluated FSAR peak pressure for the MSLB of 51.3 psig would still result in a peak pressure for the accident below the containment design pressure of 60 psig. Based on that information, screening of the condition resulted in the determination that the containment was fully operable.

Subsequently, while completing our disposition of these earlier condition reports, a preliminary containment pressure calculation was performed using sensitivity information from WCAP 15153 to account for the additional mass and energy released to containment resulting from the failure of a MFRV to close. This evaluation indicated that the containment pressure at our currently licensed rated thermal power may exceed the containment design pressure of 60 psig for a short period of time (less than seven minutes). Although the safety significance of this transient condition, as discussed in the Safety Assessment of this report, is minimal, this preliminary determination prompted the NRC notification discussed in the opening paragraph of this Event Description.

Cause:

The apparent root cause of this event was the failure to identify that the hot zero power case may not be the most limiting MSLB case with respect to containment integrity when the potential for failure of the MFRV was first identified in the 1980s. Another opportunity to address this event occurred when we evaluated WCAP 15153 and the significance of a revised containment response evaluation for an up-rated power condition. We initially assumed that the elevated containment pressure condition resulting from the failure of the main feedwater regulation valve (MFRV) to shut was the due to the higher rated power assumed for the up-rated condition. A further evaluation of the results of that analysis in 2000 incorrectly applied sensitivity information associated with the vendor analysis code and concluded that the peak containment pressure for the MSLB with failure of a MFRV would not exceed the design pressure for the licensed rated thermal power.

Corrective Actions:

In accordance with PBNP procedures and information from Generic Letter 91-18 Revision1, an operability determination for this condition was completed which concluded that the PBNP containment structures are operable but non-conforming.

We are working to determine the scope and cost of a new analysis of the containment response for the MSLB inside containment. We believe this analysis will result in a peak containment accident pressure of less than 60 psig. To achieve these results it is possible that one or more inputs to the analysis will have to be modified. We anticipate that this analysis will be completed by the end of September 2001.

After we have completed our evaluation of the revised MSLB analysis we shall supplement this LER to provide a summary of the results of that analysis.

Component and System Description:

The containment structure provides biological shielding for both normal and accident situations. The general configuration, dimensions and design aspects of the reactor containment structures at the PBNP Unit 1 and 2 are provided in Chapter 5 of the PBNP FSAR. Both containment structures at PBNP are right cylinders with a flat base slab and a shallow domed roof. The nominal 3 ft. 6 in. thick cylindrical wall and 3 ft. thick dome are prestressed and post-tensioned. The nominal 9 ft. thick concrete base slab is reinforced with high strength reinforcing steel. A ¼ inch thick welded steel liner is attached to the inside face of the concrete shell to insure a high degree of leaktightness. The base liner is installed on top of the structural slab and is covered with concrete. The containment structures for Units 1 and 2 are essentially identical. The reactor containment completely encloses the reactor and reactor coolant system and ensure that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even in the event of a gross failure of the reactor coolant system. The containment is designed to maintain leakage no greater than 0.4%/24 hours by containment air weight at a design pressure of 60 psig and 286ºF during the design basis loss of coolant accident.

Safety Assessment:

The current PBNP licensing basis analysis for the MSLB is described in FSAR Section 14.2.5, and reports the peak containment pressure to be 51.3 psig for a break occurring at hot zero power conditions. That analysis did not consider the single failure of a MFRV to close. As referred to in various places in the FSAR, and in the Technical Specifications, the design pressure of the containment structure is 60 psig. Except for Section 14.2.5, the FSAR discussions of the containment design pressure are concerned with the containment response to a LOCA, and not to the MSLB. The vendor analysis results (WCAP-15153) performed for up-rated power limits show that the peak containment pressure reaches 64.2 psig. for the MSLB accident. A recent evaluation (performed for CR 00-1304) concluded that even under the current licensed power limit, there is a possibility that the containment peak pressure may briefly exceed 60 psig during a MSLB inside containment.

The containment pressure is specifically required to remain below the design pressure after the occurrence of a LOCA for purposes of radionuclide containment. The LOCA radiological analysis assumes that the containment leakage remains below 0.4 wt% for the first 24 hours following a LOCA. This is the design bases leakage rate at a containment pressure of 60 psig. However, for the MSLB accident, the radiological analysis assumes that the steam line break occurs outside of containment, and the releases to the environment from the faulted steam generator entirely bypass the containment structure. Further, the releases assumed from the intact steam generator also bypass the containment structure via the atmospheric dump valves and/or safety valves. Thus for the MSLB analysis, any increased leakage that may occur as a result of exceeding the 60 psig design pressure has no impact on the offsite dose consequences for that accident. Since the containment structure is not credited in the MSLB radiological dose analysis, a postulated failure of the containment during this transient has no impact on the offsite or control room dose consequences.

Note that the potential small increase of the peak containment pressure above the design pressure during a MSLB with a failure of the MFRV to close, is still well within the design margin for the containment structure. As mentioned previously, the vendor MSLB analysis results for the uprated power conditions (which bound the current license conditions) indicates a peak pressure of 64.2 psig (7.5% above the design pressure of 60 psig) for the steam line break inside containment. The analysis also established that the containment pressure is predicted to remain above the design pressure for less than seven minutes. Therefore, the portion of the MSLB pressure transient that remains above the design pressure is very brief (~0.5%) compared to the design basis leakage criteria for the LOCA using a pressure of 60 psig for 24 hours. Additionally, as stated in the PBNP FSAR, at Page 5.1-21, Paragraphs 1 and 2, the containment is designed to withstand pressure loading at least 50% greater (81 psig) than those calculated for the postulated loss-of-coolant accident alone and, to withstand pressure loading at least 25% greater (67.5 psig) than those calculated for the postulated loss-of-coolant accident with a coincident design earthquake or wind. (The peak containment pressure for the LOCA is 54 psig per FSAR 14.3.4, Page 12.) Based on these considerations, an increase of even 7.5% in the design pressure for the MSLB transient should have no impact on the containment for continued operation. Accordingly, the safety significance of this event is negligible and there is no impact on the health and safety of the public or the plant staff while operating under these conditions. This event does not result in the loss of any safety functions; therefore, this event does not constitute a safety system functional failure.

Similar Occurrences:

A review of recent LERs (past three years) identified no similar events involving plant operations in a postulated unanalyzed condition which resulted in system or component parameters in excess of the design bases.