NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

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TITLE (4)
Steam Line Break with Continued Feedwater Addition

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 NAMES	DOCKET NUMBER(S)
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9)

POWER LEVEL (10)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
C. W. Fay, VICE PRESIDENT - NUCLEAR POWER	4 1 4	2 2 1 - 2 8 1 1

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

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	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 fifteen single-space typewritten lines) (16)

On August 12, 1988, Wisconsin Electric received a report from the NSSS vendor for the Point Beach Nuclear Plant which provided the results (mass and energy release rates) of a postulated steam line break accident and a post-accident containment pressure evaluation. The information in this report substantiates the concern that containment design pressure could be exceeded in a postulated main steam line break accident inside containment assuming a single failure of the main feed regulating valve to shut. The continued addition of feedwater while the main feedwater pumps discharge valves cycle shut (approximately two minutes) would result in exceeding the containment design pressure. Although the consequences of this scenario are bounded by the steam line break outside containment, modifications will be completed to provide redundant rapid acting feedwater termination in the event of a main steam line break.

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

EVENT DESCRIPTION

In February 1988, while conducting an evaluation of the safety-related scope of plant valves, a single failure scenario was postulated involving main feedwater addition during a main steam line break (MSLB) accident, which caused us to question the conclusions of our response to IE Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition." Nonconformance Report N-88-022 was written to track the resolution of this concern. As discussed in FSAR Section 14.2.5, feedwater isolation during a postulated MSLB accident is accomplished by a safety injection signal which rapidly closes the main feedwater regulating valves, trips power to the main feed pumps and closes the feedwater pump discharge valves. The latter valves are 16-inch, motor-operated gate valves which require approximately two minutes to cycle shut. The isolation of feedwater is necessary to limit the reactor coolant system cooldown and to limit the mass and energy release into the containment.

In the event of a MSLB with continued offsite AC power and assuming a single failure of a main feedwater regulating valve to close, the Point Beach design then relies upon the main feedwater pump discharge valves to isolate the feedwater flow to the faulted steam generator. As the steam generator pressure decreases during the transient, a pressure would be reached at which the condensate and/or heater drain tank pumps could begin to inject feedwater through the tripped main feedwater pumps into the faulted steam generator. This water injection would continue until the main feedwater pump discharge valves were fully closed (approximately two minutes). There are several factors to be considered, such as the time at which the pressure in the faulted steam generator became less than the shut off head of the condensate and heater drain tank pumps, the number of pumps running, and head loss in the feedwater lines. It was recognized that the amount of water injected into the faulted steam generator may exceed the amount of feedflow assumed in the plant safety analysis.

In order to more accurately assess the potential consequences of this postulated scenario, we contracted with the NSSS vendor, Westinghouse Electric Corporation, Inc. to perform a detailed reanalysis of this MSLB scenario. The above information was provided to the NRC in our letter dated March 23, 1988, which was a supplement to our response to IE Bulletin 80-04.

On August 12, 1988, we received the Westinghouse report which provided the results of this reanalysis and included the detailed mass and energy release rates. This analysis supports the initial concern that under the conditions of this postulated scenario the additional energy released to the containment would be likely to result in exceeding the containment design pressure.

SAFETY ASSESSMENT

The two main concerns in a MSLB accident are the core response, which includes a possible return-to-power situation due to excessive cooldown of the reactor coolant system, and the containment pressure response for the postulated MSLB inside containment. The latter response is dependent on the mass and enthalpy of the steam released to the containment.

Core Response

The core responses for the present cycles of operation (U1C15 and U2C14) were estimated from data in Nuclear Design Reports provided to Wisconsin Electric by Westinghouse. This evaluation concluded that the reactor cores would remain subcritical at average reactor coolant system (RCS) temperatures greater than 250°F. This evaluation was based on the actual end-of-life (EOL) shutdown margin by all-rods-in less the most reactive rod, which is assumed to be stuck in the fully-withdrawn position. These EOL shutdown margins were 3.87% and 3.96% for Units 1 and 2, respectively. The EOL case is the most severe for MSLB because the moderator temperature coefficient is the most negative at that time in the cycle.

In their generic response to NRC IE Bulletin 80-04, Westinghouse stated that the first minute of the MSLB transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal hydraulic transients in the core. It has been shown that negative reactivity inserted by concentrated boric acid from the high pressure safety injection system begins reducing core reactivity at approximately 50 seconds after the break for the analysis of MSLB inside containment in the Point Beach FSAR. Therefore, the core response is very insensitive to continued feedwater flow. The conservative FSAR analysis shows a return-to-power situation due to an EOL shutdown margin assumption of 2.77% $\Delta K/K$ and a moderator density coefficient of 0.43 $\Delta K/K/gm/cc$. These parameters are conservative compared to the expected characteristics of Point Beach fuel cycles. Therefore, the return-to-power situation should not be more severe than analyzed in the FSAR, even with the continued feedwater addition. Therefore we believe the current FSAR analyses bound this new MSLB scenario with respect to core response.

Containment Response

The pressurization of containment during a postulated MSLB inside containment occurs due to the mass and energy release into the containment. The time response of containment pressure depends upon the rate of mass and energy addition to and the rate of mass and energy removal from the containment atmosphere. The mass and energy release from the faulted steam generator depends upon the break size, steam generator pressure and enthalpy of the blowdown. The FSAR Chapter 14 MSLB analyses are initiated from the hot shutdown (HSD) condition. The maximum initial steam generator mass and energy exist for the HSD case because the mass inventory is highest at HSD. Thus, the initial mass release rate should be maximum for the HSD case. But the feedwater flow at hot shutdown is close to zero with a minimum number of condensate and heater drain tank pumps running.

For a MSLB inside containment, with the reactor operating at power, the accident analysis contained in the FSAR Section 14.2.5 states that after the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analyses which assumed a zero-power load condition at time zero. In reality, when the RCS average temperature reaches the no-load value of 547°F, the mass release rate will probably be lower than the HSD case due to lower steam generator pressure at this time in the transient. Therefore, the rate of mass release early in the transient is lower for the at-power case than for the HSD case analyzed in the FSAR.

The effect of continued feedwater addition during the initial phase of the transient is more difficult to determine. The blowdown rates early in the accident should be insensitive to feedwater addition rate. An increase in water fed to the steam generator will increase the total blowdown and provide more cooling to the primary side. After approximately one minute of elapsed time, the actuation of full containment safeguards and heat removed by containment structures will remove mass and energy from the containment atmosphere.

In the analysis performed by Westinghouse for MSLB scenario being considered, the mass and energy release rates to containment were calculated. These release rates were used by Wisconsin Electric to estimate the containment peak pressure for this scenario. The estimated peak pressure was determined to occur at approximately six minutes after the break. Based on the calculated mass and energy release, the containment design pressure of 60 psig was estimated to be exceeded.

Subsequent evaluations of the analysis and its results, however, have shown that there are assumptions and approximations that may be leading to an unrealistically high mass-energy release. The following assumptions/approximations were evaluated:

1. The Westinghouse analysis assumes that the blowdown is single-phase steam with a quality of 1.0. This assumpti[on] could under-predict the mass velocity and over-predict th[e] energy release rate. If the blowdown is actually two-pha[se,] which is likely, some mass would go directly to the containment sump. This would leave less inventory of mass available to blowdown to containment as steam. Although Westinghouse did not have information regarding this phenomena for the Model 44 steam generators at PBNP, a review of safety analysis reports for other facilities shows the use of 15% reduction in the energy release based on two-phase blowdown. This results in an approximately 15% reduction in the calculated peak containment pressure in those safety analyses that include liquid entrainment, which results in lower blowdown energy.

2. The entire feedline volume open to the steam generator was assumed to turn to steam and the heater drain tank pump suction inventory was assumed to be unlimited. It is likely that most of the feedwater in the unisolable portion of the feedwater system would not turn to steam or flow to the faulted steam generator. This would reduce the mass release by approximately 55,000 lbm. Also, the suction inventory of the heater drain tank pumps would probably be limited in this event. The heater drain tank pump discharge flow control valve shuts on a low level signal for the heater drain tank. This could reduce the total mass release by approximately 45,000 lbm. Altogether, approximately 100,000 lbm could be eliminated from blowdown in a more realistic analysis. These savings in mass inventory would substantially reduce the calculated peak containment pressure.

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3. The mass and energy release analysis assumed blowdown to a constant 14.7 psia containment back pressure. In actuality, for a MSLB inside containment, the blowdown and primary system cooldown would be limited by the pressurization of the containment building.

4. A return-to-power situation occurs in this analysis because an extremely conservative EOL moderator density coefficient and the worst-case shutdown margin are chosen. As previously explained, these worst-case conditions do not exist in the current cycles of operation and normally would not exist for Point Beach fuel cycles. Therefore, less energy would be available to blow down to containment.

Although the results of this analysis (performed by Westinghouse and Wisconsin Electric) show that the potential for containment overpressure does exist, we have concluded that the radiological consequences of the MSLB inside containment would not be more severe than those presented in the PBNP FSAR for a MSLB outside containment. These consequences are based on Technical Specification limits for fuel failure, reactor coolant activity, and primary-to-secondary leakage. The conclusion in the PBNP FSAR, which states, "No significant exposure to the public would result from a rupture of a steam pipe," remains valid for this new MSLB scenario. Therefore, continued safe operation of Point Beach is assured until the long term corrective actions are implemented.

Cause

The cause of this event is a design inadequacy which occurred during the original design of the facility.

General Implications

This event is applicable to both Point Beach Nuclear Plant Unit 1 and Unit 2. The postulated scenario which results in this design inadequacy may also be applicable to specific designs at other facilities.

Reportability

A red phone report was made on February 19, 1988, when the issue was first brought to the attention of plant personnel. This LER is provided pursuant to the provision of 10 CFR 50.73(a) (2)(v) as further clarified by Paragraph 10 CFR 50.73(a)(2)(vi). Paragraph (a)(2)(v) states that licensee shall report "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: . . . (D) mitigate the consequences of an accident." Paragraph (a)(2)(vi) states that: "Events covered in Paragraph (a)(2)(v) of this section may include. . .discovery of design, analysis, fabrication, construction, and/or procedural inadequacies."

Corrective Action

Short Term - An order has been issued to the operating personnel which instructs them in EOP-0, "Reactor Trip or Safety Injection," to trip the condensate pumps and heater drain tank pumps if a main feed regulating valve does not shut.

Long Term - Wisconsin Electric has initiated the evaluation of hardware modifications that would eliminate this scenario from consideration as a credible accident. These hardware modifications will provide the equivalent of redundant rapid termination of main feedwater flow in the event of the postulated single failure. The modification currently being studied includes automatic closure of the existing heater drain tank discharge valves and automatic tripping of the condensate pumps on a high containment pressure safety injection signal. Wisconsin Electric intends to proceed with the detailed design of this option. If the design effort does not reveal significant problems with this approach, we expect to implement the modifications during scheduled refueling outages in Fall 1989 for Unit 2 and Spring 1990 for Unit 1.

Wisconsin Electric POWER COMPANY
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VPNPD-88-460
NRC-88-086

10 CFR 50.73

September 12, 1988

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

Gentlemen:

DOCKETS 50-266
LICENSEE EVENT REPORT 88-008-00
STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION
POINT BEACH NUCLEAR PLANT, UNIT 1

Enclosed is Licensee Event Report 88-008-00 for Point Beach Nuclear Plant, Unit 1. This report is provided in accordance with 10 CFR 50.73(a)(2)(v), "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to...(D) mitigate the consequences of an accident."

This report details the discovery of a design inadequacy involving a postulated single failure during a main steam line break accident.

If any further information is required, please contact us.

Very truly yours,

C. W. Fay

C. W. Fay
Vice President
Nuclear Power

Enclosure

Copies to NRC Resident Inspector
NRC Regional Administrator, Region III

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