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AEP
INDIANA MICHIGAN POWER

October 13, 2000

United States Nuclear Regulatory Commission
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Operating License DPR-74
Docket No. 50-316

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Report System, the following report is being submitted:

LER 316/2000-013-00, "Safety Injection Valve Closure at High Power Levels Could Result in Exceeding PCT Limits."

No new commitments were identified in this submittal.

Should you have any questions regarding this correspondence, please contact Mr. Wayne J. Kropp, Director Regulatory Affairs, at 616/697-5056.

Sincerely,

A. Christopher Bakken, III
Site Vice President

/srd
Attachment

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IE22

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NRC Form 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

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

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TITLE (4)

Safety Injection Valve Closure at High Power Levels Could Result in Exceeding PCT Limits

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
09	18	2000	2000	--	013	-- 00	10	13	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	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)
POWER LEVEL (10)	100		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)
Ron Gaston, Regulatory Affairs	616 / 465-5901, x1366

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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 September 18, 2000, during a review of Unit 2 emergency core cooling system (ECCS) valve operability procedures, it was determined that a small break loss of coolant accident (SBLOCA) concurrent with a closed safety injection (SI) system containment isolation valve (CIV) at 100 percent reactor power could result in exceeding the core peak cladding temperature (PCT) limit of 2200 degrees Fahrenheit (F). A review of past operating history identified several occasions prior to Unit 2 shutdown in 1997 when, during procedural surveillance testing, Unit 2 had closed one SI system CIV at greater than 3250 MWt. Should the postulated SBLOCA have occurred during one of these periods, the 2200 degree F PCT limit could have been exceeded. A four hour non-emergency phone call to the NRC was made on September 18, 2000, in accordance with 10CFR50.72(b)(2)(iii)(D), a condition that could have prevented fulfillment of the safety function. However, reevaluation of the condition has resulted in a determination that the plant was in a condition that was outside the design basis. As such, this LER is submitted in accordance with 10CFR50.73(a)(2)(ii)(B) for a condition outside the design basis of the plant.

The root cause of the condition was a loss of design control. A programmatic root cause evaluation was performed on the design control process in 1999 that addresses the historical causes of loss of design control. The corrective actions to prevent recurrence for the root cause of the generic inadequacies of the design control process are currently being addressed through the Corrective Action Program. Surveillance procedures 2-OHP 4030.STP.053A (B), "E.C.C.S. Valve Operability Test - Train A (B)," were revised to remove the SI CIVs from the testing requirement in Operational Modes 1, 2, and 3. Technical Specifications are being evaluated for appropriate changes. An extent of condition review of ECCS identified no other related conditions.

Based on the bounding SBLOCA dose analysis, and short evolution time for cycling the SI CIVs, the safety significance of the postulated condition is minimal.

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

Conditions Prior to Event
Unit 2 was in Mode 1, Power Operation, at 100 percent Rated Thermal Power.

Description of Event
On September 18, 2000, during a review of Unit 2 emergency core cooling system (ECCS) valve operability procedures, it was determined that a small break loss of coolant accident (SBLOCA) concurrent with a closed safety injection (SI) system containment isolation valve (CIV) at 100 percent reactor power could result in exceeding the core peak cladding temperature (PCT) limit of 2200 degrees Fahrenheit (F). The postulated condition requires closure of one of the two SI system CIVs, such as during valve testing, with reactor power greater than 3250 megawatts thermal (MWt). A subsequent SBLOCA of sufficient size that results in an SI actuation signal would start the ECCS system pumps and realign systems for injection to the reactor coolant system (RCS). The SI pumps would inject through the remaining CIV to only two of the four RCS loops when RCS pressure decreased below the shutoff head for the pumps, approximately 1500 psig. The asymmetrical cooling flow into the RCS loops would result in a reduction of core cooling from the normal SI lineup to all four loops. While no specific SBLOCA analysis has been performed for this configuration, it is believed that PCT could increase to greater than 2200 degrees F due to the reduction in cooling flow. Assuming a single failure of one train of ECCS during the postulated SBLOCA, the resulting loss of one of the two SI pumps would exacerbate the condition.

A review of past operating history identified several occasions prior to Unit 2 shutdown in 1997 when, during procedural surveillance testing, Unit 2 personnel had closed one SI system CIV at greater than 3250 MWt. Should the postulated SBLOCA have occurred during one of these periods, the 2200 degree F PCT limit could have been exceeded. The condition does not exist on Unit 1 because Unit 1 is not licensed for a power level higher than 3250 MWt, and it is expected that any increase in PCT due to reduced SI flow would be accommodated within the margin to the 10CFR50.46 regulatory limit.

The condition was initially determined to be a condition that alone could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident, and was reported to the NRC in a four hour non-emergency phone call on September 18, 2000, in accordance with 10CFR50.72(b)(2)(iii)(D). However, reevaluation of the condition has resulted in a determination that the plant was in a condition that was outside the design basis. As such, this LER is submitted in accordance with 10CFR50.73(a)(2)(ii)(B) for a condition outside the design basis of the plant.

Cause of Event
The root cause of the condition was a loss of design control. A programmatic root cause evaluation was performed on the design control process in 1999 that addressed the historical causes of loss of design control. Specific causes identified in the design control root cause applicable to this condition are management expectation errors and an inadequate interface definition between functional groups. Management expectation errors resulted in unclear roles and responsibilities, low quality procedures, and excessive backlogs. Inadequate interface definition between functional groups impacted design basis and configuration management. Maintaining the design basis and providing strong configuration management are vital functions in nuclear power operations. As such, strategic errors, low expectations, and a low commitment in implementing and controlling the Donald C. Cook Nuclear Plant (CNP) design basis caused this programmatic issue.

Analysis of Event
The ECCS is the primary system that cools the reactor core and provides reactivity control to maintain the core sub-critical following breaks in the RCS or main steam system. The primary design function of the ECCS is to mitigate the consequences of any size RCS break by providing water to cool the reactor core to limit fuel clad damage. The ECCS operates during an accident in two phases. The initial injection phase commences at the receipt of a SI signal, at which time the ECCS pumps are automatically started, providing borated water to the RCS from the refueling water storage tank (RWST). Termination of the injection phase occurs before the RWST is completely empty. Continued long-term core and containment cooling are provided by recirculation of the spilled reactor coolant, RWST contents, and ice melt from the ice

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condenser pumped from the containment recirculation sump through the residual heat removal system and back to the RCS loops.

The condition of closing a SI CIV at greater than 3250 MWt potentially challenges the design function of the ECCS to maintain the reactor core cool. The postulated condition can be compared to the evaluated condition of having an SI cross-tie valve closed and experiencing a single failure that causes the loss of one SI pump, resulting in one SI pump injecting to only two of four RCS loops. CNP Unit 2 Technical Specifications (TS), section 3.5.2, "ECCS Subsystems - Tave Greater Than or Equal To 350 Degrees F," action statement [b] requires that, with a SI cross-tie valve closed, restore the cross-tie valve to the open position or reduce core power level to less than or equal to 3250 MW within one hour. This requirement is in place to ensure PCT remains within acceptable limits in the event of a SBLOCA at high power levels, assuming a single failure of one pump. The one-hour time limit in the action statement allows for the performance of periodic operability testing and system valve lineup manipulations. There is no equivalent action statement or one-hour allowance for closing the SI CIVs.

Exceeding the 2200 degree F PCT limit could cause fuel clad failure and radioactive release to the containment. The SBLOCA dose analysis is bounding for this postulated condition because the analysis assumes 100 percent cladding failure in the dose calculations. Additionally, cycling the SI CIVs for testing is a very brief evolution with the valves typically closed for two minutes or less; therefore, the concurrent SBLOCA exposure period for such a scenario is quite short. Based on the bounding SBLOCA analysis, and short evolution time for cycling the SI CIVs, the safety significance of the postulated condition is minimal.

Corrective Actions

The corrective actions to prevent recurrence for the root cause of the generic inadequacies of the design control process are currently being managed through the CNP Corrective Action Program. The root cause evaluation identified numerous corrective actions to address management, organizational, and programmatic issues in the Engineering organization.

Surveillance procedures 2-OHP 4030.STP.053A (B), "E.C.C.S. Valve Operability Test - Train A (B)," were revised to remove the SI CIVs from the testing requirement in Operational Modes 1, 2, and 3.

Technical Specifications are being evaluated for appropriate changes.

An extent of condition review was performed on the remaining ECCS containment isolation valves, and no other related conditions that would adversely impact PCT were identified.

Previous Similar Events

There have been numerous LERs from 1999 and 2000 that have been linked to the design control inadequacies root cause. Since the condition described here is historical, the corrective actions from these LERs would not have identified or prevented this condition.