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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 00-005-00

Ladies and Gentlemen:

Enclosed is Licensee Event Report (LER) 2000-005, Unrecognized Design Requirement for Emergency Service Water Resulted in Operation Outside the Design Bases
If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

John K Wood

Attachment

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22

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

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Unrecognized Design Requirement for Emergency Service Water Resulted in Operation Outside the Design Bases

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
11	01	2000	2000	5 --	00	12	8	2000		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	98.5

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)	X	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)
Kenneth F. Russell, Compliance Engineer	(440) 280-5580

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 November 1, 2000, at approximately 1300 hours, the Perry Nuclear Power Plant (PNPP), Unit No. 1, was operating at 98.5 percent rated thermal power, when it was identified that a condition had existed that resulted in operation outside of the plant's design basis. With the Emergency Service Water (ESW) alternate intake sluice gates open, recirculated discharge water would raise the ESW suction temperature above the design-input value resulting in potential inadequate cooling during a Design Basis Accident (DBA). The condition was discovered during an investigation related to opening the ESW sluice gates. A preliminary engineering calculation, being completed as a corrective action to a sluice gate issue documented in the plant's corrective action program, determined that the gates should not be opened without aligning the discharge to the surface drainage (swale). This condition has existed since the initial operation of the plant, during which the sluice gates have been opened on limited occasions to support maintenance of the gates.

The apparent cause of this condition was inadequate initial ESW design. There was no engineering design information provided by the Architect Engineer that could have allowed the understanding that opening the sluice gates at low lake temperature without realignment to the swale would result in exceeding the design inlet temperature for ESW during a DBA. This condition is a latent error that has existed since the initial operation of the plant. The condition was entered in the plant corrective action program and administrative controls were put into place to ensure the ESW system discharge is aligned to the swale prior to opening the sluice gates. Long term corrective actions will be to incorporate the final calculation results into the plants design basis. Additionally, permanent procedure controls will be implemented that reflect the design.

This License Event Report is submitted in accordance with 10 CFR 50.73(a)(2)(ii), plant outside its design bases and as a conservative measure 10 CFR 50.73(a)(2)(v), as a loss of safety function.

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

INTRODUCTION

The Perry Nuclear Power Plant (PNPP) is designed such that the normal intake tunnel supplies water to Emergency Service Water (ESW)[BS] and to non-safety Service Water [KG] from Lake Erie. In the event the intake tunnel is damaged, sluice gates [GATE] automatically open to supply water from the discharge tunnel on lowering ESW pumphouse forebay level. There is also a manual override that allows operators to control opening the gates. The ESW return water is then manually redirected to the lake via surface drainage (swale). The loss of intake tunnel is analyzed separately, and it is not required to assume a DBA occurs at the same time. If a DBA were to occur during the time when the sluice gates are open for maintenance, the non-safety Service Water system is not relied upon to operate and Service Water flow is assumed to stop. Subsequently, ESW return flow recirculates to the suction of the ESW pumps at a rate that is greater than previously understood, until operators either manually align flow to the swale or close the sluice gates.

At the time this condition was identified on November 1, 2000, PNPP was in Mode 1 at approximately 98.5 percent rated thermal power. The reactor vessel pressure was at approximately 1024 pounds per square inch gauge, with the reactor coolant at saturated conditions. There were no inoperable systems, structures or components that contributed to this condition. The License Event Report is required to be submitted to the NRC in accordance with 10 CFR 50.73(a)(2)(ii), since the condition was outside the plant's design basis and as determined from the ensuing evaluation conservatively under 10 CFR 50.73(a)(2)(v), as a loss of safety function. Additionally, on November 30, 2000, the NRC was notified via telephone that the LER report date needed to be extended to support additional significance evaluation.

EVENT DESCRIPTION

Several Engineering initiatives have been ongoing as a result of recent elevated Ultimate Heat Sink (UHS) temperature concerns. A Condition Report written in April 2000 identified a correlation between the UHS temperature and the need to establish an upper temperature limit to allow opening the sluice gates. A rigorous calculation by an outside contractor was performed to evaluate the lake temperature at which the sluice gates could be opened while maintaining the cooling capability of safety systems. The preliminary results of the computer-generated calculation, received on November 1, 2000, determined that the ESW discharge should be realigned to the swale any time the sluice gates are open. The analysis determined that recirculated water eventually causes the inlet temperature to ESW to exceed the design input value of 85 degrees Fahrenheit under worst case accident conditions without reliance on operator action. This could potentially result in ESW not being able to perform the design function of removing heat from the DBA required loads and decay heat. The primary load during the DBA scenario is the Residual Heat Removal (RHR)[BO] system. The reduced cooling capability of RHR affects removal of decay heat from the reactor core, suppression pool and containment. Other loads supplied by ESW that would be challenged are Divisional Diesel Generator [EK]Cooling, Fuel Pool Cooling [DA] and Emergency Closed Cooling [BI] and its associated loads.

Contrary to this most recent understanding, the sluice gates have been opened on limited occasions during the operating history of the plant. The primary reason for opening the sluice gates was in order to perform periodic maintenance. For the majority of the occasions when a sluice gate was open, an operator would have been in the ESW pumphouse and would have closed the gates had an event occurred. On eight separate known occasions, (March 2000, about 24 hours; March 1999, about 16 hours; March 1998, about 1 hour; April 1998, about 6 hours; May 1995, less than 72 hours; November 1991, less than 24 hours; February/March 1990, less than 72 hours; September 1987, less than 72 hours), a gate was removed or incapable of being closed by it's safety-related motor and would have required additional effort to reinstall. On these separate, but limited occasions when the sluice gates could not be easily closed, operator action would have been required to align the discharge to the swale.

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

CAUSE OF EVENT

The apparent cause of this condition was inadequate ESW design. The full understanding of the ESW design was not known until the recent calculation was performed. The calculation was performed in response to an ongoing corrective action program investigation of sluice gate bypass leakage, to determine its impact on the UHS temperature due to elevated temperatures in Lake Erie over the previous year(s). During this investigation the NRC Senior Resident at the plant questioned the maximum temperature at which the gates had been opened. When it was determined that gate opening had occurred at higher temperatures than previously thought to be the limit, a rigorous calculation was requested to determine the maximum temperature that the gates could be opened without necessitating realignment to the swale. The preliminary results of this calculation presented information that indicated the ESW should be pre-aligned to the swale anytime the sluice gates were open. Since this information was not previously known, the design was inadequately documented within the plant's design basis, and therefore not properly reflected in the plant's operating procedures. This condition has existed since the initial design of the plant and is considered a latent design error.

SAFETY ANALYSIS

The safety significance of this condition is determined by the ability of the plant operators to take reasonable, non-heroic actions to restore the ESW system configuration and reestablish full capability of the (UHS). The factors affecting this capability are the time needed to recognize the appropriate action, whether the response can be performed in the time required to prevent adverse consequences and by radiological conditions.

During the majority of occasions when the sluice gates have been open for testing, preventative maintenance, or for access for diver inspection/silt removal in the tunnels, safety-related power was available. Operators were typically in the ESW pumphouse, the location of the sluice gates, and would have observed the ESW pumps and other support equipment starting if a DBA had occurred. It is expected that the operator would have communicated this information to the control room staff. For diving activities, radio communications were required between the lead diver in the pumphouse and the control room staff. During the majority of the work associated with sluice gates, maintenance personnel were in the ESW pumphouse. Although information would be readily provided to the control room to recognize the need for appropriate action, it could not be assured that continuous manning of the ESW pumphouse was maintained.

Timely recognition that action was required to ensure full integrity of the UHS in most cases would be prompted by communications from the ESW pumphouse at the start of a DBA. The communications would augment control room knowledge of plant status. The status of the sluice gate, if open, would have been documented on the control room staff's turnover sheets if the sluice gate was being maintained open between shifts due to on-going work activity. However, most of the activities were of short duration and would have been directly authorized by the control room staff. Additional recognition is provided by several informational status indications that are located on the Emergency Core Cooling System (ECCS) Benchboard in the control room. These indications include alarms that annunciate if high temperatures are encountered, an amber inoperable light for each sluice gate, open/closed indicating lights next to the ESW pump control switches and ESW temperature indication next to the ESW flow indication. These indications would be observed at the beginning of each shift during procedurally required observations of control room operating panels. In addition, if a DBA had occurred, operators are required and trained in the simulator, to observe control room panel indications to ensure accident mitigation equipment is functioning as required. The proximity of the indications to significant accident mitigation equipment (i.e. the ESW pump control switch) would further ensure that the control room staff would recognize and take appropriate action if it had existed during this event.

Timely corrective action to maintain the UHS would be considered a high priority once the control room staff became aware that a temperature limit was being approached or being exceeded. Maintaining the UHS available is fundamental operator knowledge of nuclear safety and is reinforced by loss of heat sink accident scenario drills in the plants control room simulator. It is therefore reasonable to conclude that the condition would have been identified by the control room licensed operators and corrective actions would have been directed.

Closing the sluice gate, if directed, would be accomplished from the ESW pumphouse. An operator, if not already in the pumphouse to support sluice gate operations, could be dispatched in a short time. Radiation levels during a DBA do not need to be considered for this condition. However, realignment of the swale discharge valves does require dose consideration by the operator.

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The elevated dose rates were determined to be within reasonable radiation levels to permit access to the ESW discharge valves. A conservative estimate of the radiation levels was made using the NRC approved analysis as documented in Regulatory Guide (RG) 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Applying this methodology, the dose to the operator would be within limits to perform the evolution given the expected duration and a conservative start time of 30 minutes post DBA.

Furthermore, the above operator dose conservatively assumes actual core damage during the accident. However, Perry specific 10CFR50.46 and Appendix K ECCS LOCA analysis performed using the SAFER/GESTR methodology indicate that the maximum expected peak cladding temperature (PCT) is less than 1400°F as shown in USAR Section 6.3. This ECCS analysis considered the worst-case pipe break, fuel type, and single active failures. Since the calculated PCT is substantially less than the 2200 degree Fahrenheit (F) regulatory limit, actual failure of the fuel at Perry during a worst case DBA accident is not expected. As such, the actual radiation levels/dose rates associated with the operator action to access and manually close the ESW discharge valves would be minimal.

In order to determine if adequate time was available to perform mitigation activities, an analysis was performed of the effect on ESW as a result of an open sluice gate. The engineering analysis of this condition identified that the increased ESW inlet temperature would be delayed beyond the start of an accident. The major heat load occurs when the RHR heat exchanger bypass valve is closed to place the heat exchanger in the cooling mode of operation. This is prevented by interlock for 10 minutes following the start of a DBA. It is expected that ESW inlet temperature will not change appreciably for an additional 20 minutes under full recirculation conditions. This delay results from more than 490 feet of 10-foot diameter tunnel piping, and from the volume in the ESW forebay that must be pumped through the system prior to the warmer recirculated ESW water entering the ESW suction. In addition, during a DBA, the flow rates are lower than normal since the service water pumps are not running and the ESW pumps, which are approximately half the capacity of the service water pumps, are running. Therefore, ESW inlet heatup from the accident heat load is not expected for approximately 30 minutes. Review of the sluice gate history indicated the evolutions that would require additional restoration efforts were normally performed in the spring or fall when the lake temperature was less than 50 degrees F. Assuming a conservative heatup, it would require from 30 to 60 minutes following the DBA for the ESW inlet temperature to reach 85 degrees F. By staying under 85 degrees F, the continued operability and thus the safety function of ESW is assured. Based on the time required to restore ESW and the time available at the start of the accident (before ESW inlet temperature is significantly impacted), it is reasonable to conclude adequate time would have been available to mitigate this condition during a DBA.

Evaluation of this issue from the Probabilistic Safety Analysis perspective indicates a potential core damage path as a result of a large LOCA event. However, the probability of occurrence of a significant issue arising from this issue is considered to be remote on the basis of the frequency of a relevant initiating event occurring. Collective industry data, likewise, supports this contention. Additionally, the functionality of the ECCS injection systems would be expected such that immediate failure would not be imminent, thus providing additional time for actions to be taken as discussed above. An estimate of incremental conditional core damage probability (ICCDP) based on the time frame involved for a individual worst case occurrence, and assuming no credit for operator actions, it is qualitatively estimated that the ICCDP would be on the 10E-07 order of magnitude. Therefore, from the PSA perspective, the conclusions drawn above appear to be reasonable.

Energy Industry Identification System (EIIS) Codes are identified in the text by square brackets [XX].

SIMILAR EVENTS

LER 1999-001 describes a condition not previously recognized that the Control Complex structural walls were not adequately designed to withstand tornado loads. This condition was identified during the review of an Operating Experience report from another plant and was an original plant design issue. Although a similar design issue, the corrective actions from this LER could not reasonably have been expected to have prevented this condition.

CORRECTIVE ACTIONS

The condition was entered into the plant corrective action program. Administrative controls were put in place to ensure the ESW system discharge is aligned to the surface discharge (swale) prior to opening the sluice gates.

Long term corrective actions will be to incorporate the final calculation results and into the plants design basis. Additionally, permanent procedure controls will be implemented that reflect the design.