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Everett P. Perkins, Jr.
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2000-0068
A4.05
PR

May 15, 2000

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Reporting of Licensee Event Report

Gentlemen:

Attached is Licensee Event Report (LER) 00-004-00 for Waterford Steam Electric Station Unit 3. This report provides details of a situation in which the Feedwater Isolation Valves may have closed faster than assumed in the design basis. This condition is being reported pursuant to 10CFR 50.73(a)(2)(ii)(B) as a condition that was outside the design basis of the plant.

There are no commitments contained in this submittal. If you have any questions concerning this LER, please contact G. Chris Pickering at (504) 739-6256.

Very truly yours,

A handwritten signature in black ink that reads "Everett P. Perkins, Jr." with a long horizontal flourish underneath.

E.P. Perkins, Jr.
Director,
Nuclear Safety Assurance

EPP/GCP/rtk
Attachment

JE22

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cc: E.W. Merschoff, (NRC Region IV)
N. Kalyanam, (NRC-NRR)
A.L. Garibaldi
P. Lewis - INPO Records Center
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NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division

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 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)
Waterford Steam Electric Station, Unit 3

DOCKET NUMBER (2)
05000-382

PAGE (3)
1 of 5

TITLE (4)
Potential Degradation of Feedwater Isolation Valves due to Design Basis Deficiencies

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
04	13	00	00	004	00	05	15	00	N/A	N/A
									N/A	N/A

OPERATING MODE (9)	POWER LEVEL (10)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)			
1	100	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(2)(i)	20.2203(a)(3)(i)	X	50.73(a)(2)(ii)
		20.405(a)(1)(ii)	20.2203(a)(3)(ii)		50.73(a)(2)(iii)
		20.2203(a)(2)(ii)	20.2203(a)(4)		50.73(a)(2)(iv)
		20.2203(a)(2)(iii)	50.36(c)(1)		50.73(a)(2)(v)
		20.2203(a)(2)(iv)	50.36(c)(2)		50.73(a)(2)(vii)
					OTHER Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME
G. Chris Pickering / Licensing Engineer

TELEPHONE NUMBER (Include Area Code)
(504) 739-6256

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 EPIX

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE). **X** **NO**

EXPECTED SUBMISSION DATE (15)

MONTH DAY YEAR

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

On 3/21/00, with Waterford 3 operating at 100% power, it was identified that both Feedwater Isolation Valves (FWIVs) may have closed faster than the 1.5 second design basis limit. A new calculation methodology and the latest stroke time data were used to make this determination.

On 4/13/00, an evaluation determined that within the last two years there were seven occasions in which the percent increase in fast valve closure load caused by waterhammer from quicker stroke times exceeded the values provided to maintain operability. These forces may have exceeded the capability of piping supports, which could have resulted in the subsequent loss of the containment isolation function. This is reportable under 10CFR50.73(a)(2)(ii)(B) as a condition that was outside the design basis of the plant. Based on an evaluation of the present conditions and immediate actions, it was determined that the FWIVs and their associated penetrations could perform their required safety functions. The root cause of this condition was Entergy's acceptance of vendor analysis without adequate review and insufficient awareness of the impact of actions on safety/reliability. Corrective actions are being addressed under the plant corrective action program. This event did not compromise the health and safety of the public.

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

REPORTABLE OCCURRENCE

On 3/21/00, Entergy determined that both Feedwater Isolation Valves (FWIVs) may have closed faster than the 1.5 second design basis limit. On 4/13/00, a new analytical methodology showed that there were instances within the last two years in which the valves could have closed faster than the minimum closure time allowed. This condition could have caused an increase in the fast valve closure (FVC) load placed on the piping due to a possible increase in waterhammer. Six instances for valve FW-184A and one instance for valve FW-184B were identified that could have caused the valves to close at speeds capable of generating forces that the supports may have been unable to withstand. This condition was then determined to be reportable under 10CFR50.73 (a)(2)(ii)(B) as a condition that was outside the design basis of the plant.

INITIAL CONDITIONS

At the time of discovery of the past events, Waterford 3 was operating in Mode 1 at approximately 100% reactor power. No structures, systems or components were out of service that contributed to this event.

EVENT DESCRIPTION

In 1993 Entergy contracted Anchor/Darling to provide "engineering and assistance" with a modification to the Feedwater Isolation Valves [ISV] (FWIVs), FW-184A(B), to replace the existing 8.5 gallon accumulators with 11.0 gallon accumulators with stop tubes. Entergy specified the valve closure must be between 1.5 and 5 seconds with either one or two accumulators in service. During Refuel 6 in 1994, Anchor/Darling conducted on-site tests of an actuator coupled to a test rig designed to simulate valve loads during varying system conditions. The data used to complete the calculations included the Anchor/Darling test results and actual valve/actuator stroke times measured by Waterford 3 personnel. Anchor/Darling provided test report CTS-26, which indicated a single actuator, no-load closure time acceptance criteria of 1.75 to 2.75 seconds. The Anchor/Darling results were used to

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complete Revision 5 to DC-3364, which incorporated the new acceptance criteria. On 3/06/98, CR-WF3-98-0337 documented that the Design Basis Review (DBR) determined the maximum differential pressure at the FWIVs was higher than initially assumed. It also determined that both accumulators were needed for the FWIVs to close against this higher pressure. On 3/16/98, DBR Open Item OI-FW-077 documented that CTS-26 used a non-conservative friction factor. On 6/21/98, CR-W3-98-0854 documented fluctuations in accumulator pressures on FW-184A(B). Corrective Action 002 of this CR required re-analysis to determine the effect of increased accumulator pressures on closure time. On 3/21/00, CR-WF3-2000-0249 documented, based on the new calculation methodology and current In-Service Testing stroke time data that the FWIVs may have closed faster than the design basis minimum limit. An initial operability evaluation was conducted pursuant to Waterford 3 procedure W4.101, which determined the valves were operable and capable of performing their safety functions. This was based on an engineering evaluation that determined, based on current conditions, the faster closure of the FWIVs would not result in a FVC load that would prevent the FWIVs and their associated penetrations from performing their required safety functions. A subsequent evaluation was performed to determine if at any time in approximately the last two years the increase in FVC load may have exceeded the allowable loads determined by the engineering evaluations. On 4/13/00, it was determined that there were seven instances when the valves could have closed at a rate that would potentially cause damage to pipe supports. These instances included six occasions for FW-184A and one occasion for FW-184B.

CAUSAL FACTORS

Casual factors for the condition include:

1. The acceptance of Anchor/Darling's closure time analysis document without a rigorous questioning attitude and an analytical basis by Waterford 3 personnel.
2. Insufficient awareness of the impact of actions on safety/reliability.

The vendor's primary objective communicated in their correspondence was to meet the 5 seconds

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maximum closure time. The 1.5 seconds limit was considered a secondary concern. The vendor described a quicker valve closure as "conservative." A valve that closes quicker than anticipated is conservative in protecting the 5 seconds limit, but would be non-conservative in protecting the 1.5 seconds limit. The vendor report failed to address this issue.

CORRECTIVE ACTIONS

Immediate Actions:

Entergy entered procedure W4.101, Operability Confirmation Process, to confirm the operability of the FWIVs and investigate the potential consequences of the quicker closure times. This process used an average of the most recent Inservice Testing measurements to re-analyze the scenario. These no-load average closure times were used as input to the analytical methodology to determine the two accumulator design basis accident load closure times. The new times were used to determine the percent increase of the FVC load. These increased loads were then applied to the piping and piping support stress calculations to determine if the piping and piping supports could withstand the increased load.

Other corrective actions are being addressed through the Waterford 3 corrective action program.

SAFETY SIGNIFICANCE

The Main Steam Line Break (MSLB) and the Feedwater Line Break (FWLB) events are the only ones of concern from Chapter 15 of the UFSAR. Both of these events have the potential to increase the feedwater mass flow through the FWIV on the affected Steam Generator (SG) and consequently create the greatest waterhammer loads. Considering the increased waterhammer loads, the FWLB radiological consequences are bounded by the MSLB. The MSLB event radiological consequences are conservatively analyzed assuming that the break occurs outside of containment and that the affected SG blows down directly to the atmosphere. The predicted MSLB event radiological consequences are

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within 10CFR100 limits. Thus, containment integrity does not affect the results of the MSLB event's radiological consequences and the possibility of containment integrity loss due to excessive waterhammer loads remains bounded with respect to dose consequences.

This event is not considered a Safety System Functional Failure since the postulated waterhammer event does not prevent emergency feedwater to both steam generators. One steam generator remains in service.

SIMILAR EVENTS

LER 99-014-00 provides details of a reactor shutdown due to a loss of Reactor Coolant Pump controlled bleed-off flow. LER 99-011-00 provides details of a reactor shutdown due to loss of Controlled Bleed-off flow. Both reports involved an actuation of an Engineered Safety Feature or the Reactor Protection System, and they were both caused by the failure of the rotating baffle of the Reactor Coolant Pump. One of the causal factors involved in the failures was inadequate OEM review of their design change.

LER 98-006-00 addresses the discovery of the FWIVs being inoperable in excess of the Technical Specification allowed outage time. The FWIVs were determined to require both accumulators for service. Prior to this event, it was believed that only one actuator was required. Therefore, one accumulator could have been out of service, and the valve not considered inoperable. Technical Specification 3.6.3 requires the plant to enter a four hour ACTION statement with an inoperable isolation valve. The use of a non-conservative valve factor during original design was a causal factor in this event.

ADDITIONAL INFORMATION

Energy Industry Identification System (EIIIS) codes are identified in the text within brackets [].