NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit Two	05000 270	1 OF 6

TITLE (4)
Containment Isolation Valve Technically Inoperable Due To Deficient Design Analysis

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(S)
										05000
12	11	96	96	- 06	- 02	07	16	97		05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)

OPERATING MODE (9)	N		
POWER LEVEL (10)	0		
[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 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)	[X] 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)	
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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 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)

In December 1995, a NRC inspection of Oconee's implementation of Generic Letter 89-10 identified an issue with Anchor/Darling double disc gate valves. As a result, a Duke calculation using EPRI methodology concluded that the valve's disc wedges have to be installed in a preferred direction. During the recent shutdown of all three Oconee Nuclear Station units, an inspection of these type valves identified four containment isolation valves that had the wedges installed in the non-preferred direction. These valves were determined to be technically inoperable from the date of installation. Further engineering analysis completed on July 10, 1997, concluded that three of the four valves had been technically operable in the past. The root cause of this event is determined to be a deficient Design Analysis, Unanticipated interaction of components. Corrective actions included reassembly with the disc lower wedges in the preferred direction and revising applicable valve maintenance procedures to provide proper guidance on installing wedges in the preferred direction during reassembly.

BACKGROUND

2FDW-103, 3FDW-103 and 2FDW-104 are motor operated valves that are the outside containment penetration isolation valve for the Steam Generator shell drain lines. During normal operating conditions these valves remain closed and are only opened during startups and shutdowns.

2HP-20 is a motor operated valve that is an inside containment isolation valve for the Reactor Coolant seal return line. During normal operating conditions this valve is in the open position.

In the event of a design basis accident, the Engineered Safeguard System [EIIS:JE] will automatically close these valves and block the operation of the opening circuit on low Reactor Coolant System [EIIS:AB] pressure or high Reactor Building pressure.

EVENT DESCRIPTION

Late in 1994, EPRI issued a report on Motor Operated Valve Performance Prediction Methodology.

Oconee Nuclear Station (ONS) had a vendor perform valve analysis using portions of the EPRI methodology. This analysis showed that Anchor/Darling double disc gate valves could meet the thrust prediction for theoretical flow isolation. The orientation of the valve disc used in this analysis was believed to be more severe than was applicable at ONS.

Between December 4 and 15, 1995, the NRC performed an inspection to verify that ONS met the intent of Generic Letter 89-10 (Safety-Related Motor Operated Valve and Surveillance Program). During this inspection, the NRC requested that ONS use the EPRI methodology to address hard seat contact on Anchor/Darling double disc gate valves. This was identified as an Inspector Follow-up Item (IFI). As a result of the IFI, ONS Engineering initiated a formal calculation as part of the design basis documentation for Anchor/Darling double disc gate valves.

On March 15, 1996, the NRC issued a Safety Evaluation Report on the EPRI methodology report, which basically approved its use as "state-of-the-art" for valve analysis.

The ONS Engineering calculation was approved on July 22, 1996, and concluded that these valves' lower wedges are required to be installed toward the downstream side (the preferred direction) except for Unit 1, 2, and 3's SSF-97 which can perform its intended function in the non-preferred direction. As a result, work orders were initiated to inspect all affected Anchor/Darling double disc gate valves during the next outage of sufficient length.

On November 12, 1996, with Unit 3 shutdown for a Refueling Outage, an inspection of 3FDW-103 identified that the wedges were installed in the non-preferred direction. As a result, a Problem Investigation Process (PIP) report was initiated and an operability evaluation began.

On November 26, 1996, with Unit 2 shutdown due to a forced outage, an inspection of 2HP-20 identified that the wedges were installed in the non-preferred direction. As a result, another PIP report was initiated and an operability evaluation began.

On December 3, 1996, the wedges on 2HP-20 were removed and reinstalled in the preferred direction.

On December 4, 1996, the wedges on 3FDW-103 were removed and reinstalled in the preferred direction.

On December 11, 1996, it was concluded that 3FDW-103 would not have been able to achieve hard seat contact since it was installed on December 9, 1989. Therefore, it was conservatively concluded that it could not close leak tight and provide containment isolation following an Engineered Safeguards actuation.

On December 23, 1996, the operability evaluation concluded that 2HP-20 had been technically inoperable since it was installed on March 5, 1992. Therefore, the valve may not have been capable of providing containment isolation during an Engineering Safeguards actuation.

On January 4, 1997, the wedges on 2FDW-103 and 2FDW-104 were inspected and found to be in the non-preferred direction. The wedges were removed and reinstalled in the preferred direction.

On January 6, 1997, it was concluded that 2FDW-103 and 2FDW-104 would not have been able to achieve hard seat contact since their installation on April 1, 1988. Therefore, it was conservatively concluded that they could not close leak tight and provide containment isolation following an Engineered Safeguards actuation. Further research into the licensing basis revealed that 3FDW-103, 2FDW-103 and 2FDW-104 are outside isolation valves in a seismic system with a closed loop inside containment. In accordance with a NRC Safety Evaluation Report dated November 11, 1981 on testing requirements of 10CFR50, Appendix J, these valves are not required to perform a containment isolation function, and therefore any failure to achieve hard seat contact is not reportable. On July 10, 1997, Engineering concluded that 3FDW-103, 2FDW-103 and 2FDW-104 were past operable for the functions they are required to perform.

All of the other Anchor/Darling double disc gate valves were inspected and found to be in the preferred direction.

CONCLUSIONS

The root cause of this event is determined to be a Deficient Design Analysis, unanticipated interaction of components. At the time this valve was installed, the vendor indicated that there was not a preferred direction. Therefore, it was not known that the Anchor/Darling double disc gate valves had to be installed with the wedges in a preferred direction. This was not clear until the completion of the EPRI testing. It is concluded that if this information had been known at the time this valve was installed, the preferred direction of the wedges could have been properly ensured.

This event is considered to be non-recurring. No corrective action can be taken to assure that industry operating experience, improving technology, testing methods, and/or analytical models do not reveal previously unknown equipment problems. Duke Power's Operating Experience Program is intended to assure evaluation of new industry information for potential impact on equipment/systems at Duke's Nuclear Power Stations.

This event did not involve an equipment failure and is not NPRDS reportable. There were no radiological overexposures, radioactive releases, or personnel injuries associated with this event.

CORRECTIVE ACTIONS

Immediate

None

Subsequent

1. All applicable Anchor/Darling double disc gate valves were inspected to ensure that the wedges were in the preferred direction.

2. Wedges on affected valves were placed in the preferred direction.

3. All applicable procedures have been revised to provide proper guidance on installing wedges in the preferred direction during reassembly.

Planned

None

Because correction of the valve disc configuration is complete and there are no planned corrective actions, none of the corrective actions contained in this report are considered an NRC commitment.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

SAFETY ANALYSIS

The Anchor Darling valve in this event is a containment isolation valve. The safety significance of this event is the potential for failure of the affected valve to close on demand following an accident. Postulating failure of these valves due to this condition, with an assumed single failure of the remaining containment boundary, could result in a postulated uncontrolled release in excess of the allowed containment leakage limits.

2HP-20 is the inside containment isolation valve in the reactor coolant pump seal return line. In order to lose containment integrity following an accident, 2HP-21, the outside containment isolation valve, would also have to fail to close. 2HP-21 has a pneumatic operator, which fails closed on loss of air.

The potential failure of valve 2HP-20 does not increase the probability of any PRA core damage sequence (such as LOCA outside of containment). This is because the High Pressure Injection [EIIS:BG] makeup system could keep up with any flow out of 2HP-20.

As stated above, 2HP-21 could be used to isolate any containment leakage through 2HP-20.

There were no releases of radioactive material involved with this incident. The health and safety of the public were not affected.