FENOC

FirstEnergy Nuclear Operating Company

5501 North State Route 2
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Lew W. Myers
Chief Operating Officer

419-321-7599
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Docket Number 50-346

License Number NPF-3

Serial Number 1-1338

November 26, 2003

Mr. James L. Caldwell, Administrator
United States Nuclear Regulatory Commission, Region III
801 Warrenville Road
Lisle, IL 60532-43551

Subject: Closure of Confirmatory Action Letter Item 3

Dear Mr. Caldwell:

As a result of the identification of degradation of the Davis-Besse Nuclear Power Station (DBNPS) reactor pressure vessel (RPV) head in March 2002, the Nuclear Regulatory Commission (NRC) issued FirstEnergy Nuclear Operation Company (FENOC) Confirmatory Action Letter (CAL) 3-02-001 on March 13, 2002, to document the six specific activities FENOC intended to implement to resolve the RPV head degradation issue. Item number three of the CAL was listed as:

(3) Evaluate and disposition the extent of condition throughout the reactor coolant system relative to the degradation mechanisms that occurred on the RPV head.

On August 16, 2002, the NRC transmitted to FENOC the Restart Checklist of issues that require disposition prior to restart in accordance with the Inspection Manual Chapter 0350 process. Restart Checklist Item No. 2.c addressed the Adequacy of Structures, Systems, and Components Inside Containment.

On July 26, 2002, the NRC completed a special inspection at the DBNPS to review the actions to resolve Restart Checklist Item No. 2.c, focusing on review of activities as described in the Containment Boric Acid Extent of Condition Plan (NRC Inspection Report Number 50-346/02-09(DRS) dated September 13, 2002). Due to issues discovered with implementation of the Extent of Condition Plan, Restart Checklist No. 2.c remained open at the end of this inspection. Additional inspection activities were completed on October 24, 2002, regarding the Boric Acid Corrosion Extent of Condition (NRC Inspection Report Number 50-346/02-12(DRS) dated November 29, 2002). While this inspection concluded the Containment Health Assurance Plan was effectively implemented, three unresolved items (URIs) associated with corrective actions on

DEC 1 2003

components potentially affected by boric acid corrosion were identified. These three URIs were as follows:

- 50-346/02-12-01, Potential Leakage at the Reactor Vessel Incore Penetration Tubes
- 50-346/02-12-02, Potential Impact of Corrosion on the Ground Function of Electrical Conduit in Containment
- 50-346/02-12-03, Potential Failure to Follow the Procedure for Raychem Splice Removal on Electrical Cables

A brief summary of each unresolved item, as well as the corrective actions taken to address each item is included in Attachment 1 of this letter.

Recent inspection activities at the DBNPS included the Corrective Action Team Inspection (CATI) and the observation of the Normal Operating Pressure Test of the Reactor Coolant System. During these inspection activities, the NRC inspectors conducted a review of the corrective actions that FENOC has taken for each of these unresolved items, and information on each was provided to the inspectors to facilitate review and closure. Additionally, the closure documentation for each item has been captured in the DBNPS Regulatory Commitment Tracking System (RCTS).

Per discussions with your staff, these three unresolved items are the last remaining open items associated with CAL Item #3 to evaluate and disposition the extent of condition throughout the reactor coolant system relative to the degradation mechanisms that occurred on the Reactor Pressure Vessel head. Since these items have been inspected, FENOC respectfully requests that these items, as well as CAL Item #3, be closed.

There are no new regulatory commitments contained in this letter. If there are any questions concerning this matter, please contact Mr. Kevin L. Ostrowski, Manager – Regulatory Affairs at 419-321-8450.

Sincerely,

GMW/s

Attachments

cc: U.S. NRC Document Control Desk
John A. Grobe, Chairman NRC 0350 Panel
DB-1 Senior NRC/NRR Project Manager
DB-1 Senior NRC Resident Inspector
Utility Radiological Safety Board

Summary of Unresolved Items from NRC Special Inspection Report 50-346/02-12(DRS)

URI 50-346/02-12-01, Potential Leakage at the Reactor Vessel Incore Penetration Tubes

As documented in the DBNPS response to NRC Bulletin 2003-02 (Serial Letter Number 2992 dated 11/19/2003) as well as Licensee Event Report 2002-007 Revision 01 (dated 11/14/2003), an inspection of the incore penetration tubes was performed by raising the Reactor Coolant System to normal operating pressure using non-nuclear heat and maintaining for approximately seven days. Pressure was then reduced and a bare metal visual inspection for evidence of leakage was performed. No new boric acid deposits or other indications of leakage were identified on the reactor vessel incore penetration tubes.

URI 50-346/02-12-02, Potential Impact of Corrosion on the Ground Function of Electrical Conduit in Containment

This unresolved item was evaluated under the DBNPS Corrective Action Program as Condition Report (CR) 2002-06788. The evaluation concluded the boric acid on the conduits caused neither an electrical or structural problem. The conduits at the DBNPS are not used as, nor are they designed to be, a credited neutral conductor for either three-phase or single-phase circuits. No conduit or conductor was degraded to the point where the electrical continuity was adversely affected.

URI 50-346/02-12-03, Potential Failure to Follow the Procedure for Raychem Splice Removal on Electrical Cables

This unresolved item was evaluated under the DBNPS Corrective Action Program as CR 2002-05459. The evaluation concluded that damage of the electrical cable was caused by incorrect worker practices during the removal of the Raychem splices. This work was performed by contract electrician personnel hired specifically for work on the Containment Air Coolers. Subsequently, the three power cables to the Containment Air Cooler fan motors have been replaced, and training for electrical contractors has been amended to prevent recurrence. Additional corrective actions associated with contractor control were addressed by CR 2002-05991.

Further details regarding the investigations/evaluations performed and the specific actions taken for each item are available upon request.

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station in this document. Any other actions discussed in the submittal represent intended or planned actions by Davis-Besse. They are described only as information and are not regulatory commitments. Please notify the Manager – Regulatory Affairs (419-321-8450) at Davis-Besse of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
None	N/A

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NP-33-03-001-02

Docket No. 50-346

License No. NPF-3

November 26, 2003

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Ladies and Gentlemen:

LER 2003-001-02
Davis-Besse Nuclear Power Station, Unit No. 1
Date of Occurrence – January 30, 2003

Enclosed please find Supplement 2 to Licensee Event Report 2003-001, which was submitted to provide written notification and extent of condition of potential inability of air-operated valves to function during some design basis conditions. This issue was identified during development and implementation of an Air-Operated Valve Reliability Program as previously committed in Licensee Event Report 2002-004. This Supplement addresses risk significance of the identified valve conditions and provides commitment updates and administrative corrections. Commitments associated with this LER are listed in the Attachment. This issue is being reported pursuant to 10CFR50.73(a)(2)(i)(B), an operation or condition prohibited by the plant's Technical Specifications.

Very truly yours,

PSJ/s

Enclosures

cc: Regional Administrator, USNRC Region III
DB-1 Project Manager, USNRC
DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

COMMITMENT LIST

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COMMITMENTS	DUE DATE
1. Install sufficiently sized accumulators for valves CC1495, SW1424, SW1429, and SW1434.	1. Completed
2. Install high pressure nitrogen bottles for valves SW1357 and SW1358.	2. Completed
3. Install a high pressure nitrogen bottle for valve SW1356 and perform post-modification testing.	3. Prior to restart.
4. Install new valve actuator and accumulator capable of closing MU3 against the maximum RCS differential pressure.	4. Completed
5. Submit LER Supplement to address risk significance.	5. Completed
6. Continue implementation of Design Interface Evaluation Program.	6. Ongoing
7. Continue implementation of AOV Reliability Program to ensure maintenance of AOV design adequacy.	7. Ongoing

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104	EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

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

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Potential Inability of Air-Operated Valves to Function During Design Basis Conditions

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
01	30	2003	2003	-- 001 --	02	11	26	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	D
10. POWER LEVEL	000

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Peter S. Jordan – Regulatory Affairs	(419) 321-8260

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On January 30, 2003, with the reactor defueled, it was determined that several air-operated valve (AOV) actuators had negative operating margins. After further review, it has been determined that eight valves in total were not capable of performing their intended safety functions for the most limiting conditions. Valve MU3 would not fully close against maximum Reactor Coolant System pressure to isolate letdown flow. A new actuator and accumulator has been installed. Upon a loss of non-safety instrument air, valves CC1495, SW1356, SW1357, SW1358, SW1424, SW1429, and SW1434 would not have been capable of performing their intended safety functions because they were not provided with sufficiently sized accumulators. Following some postulated accidents, CC1495 is to close to isolate Component Cooling Water (CCW) flow to non-essential equipment. SW1356, SW1357, and SW1358 may be remote manually closed to provide containment isolation function for Service Water (SW) return from non-operating Containment Air Coolers. SW1424, SW1429, and SW1434 are to open to ensure adequate SW flow through the CCW heat exchangers. Sufficiently sized accumulators are being installed on these seven valves. These conditions, apparently caused by design assumptions that were not fully adequate, are being reported in accordance with 10CFR50.73(a)(2)(i)(B) as operation or condition prohibited by the Technical Specifications.

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

DESCRIPTION OF OCCURRENCE:

System and component design basis calculations have been performed and reviewed as part of the development and implementation of an AOV Reliability Program as previously committed in Davis-Besse Nuclear Power Station (DBNPS) Licensee Event Report 2002-004. On January 30, 2003, with the reactor defueled, it was determined that four air-operated valve (AOV) actuators had negative operating margins in some limiting conditions. After initially reporting the four deficient valve actuators, a total of eight valve actuators was determined to be deficient. Upon loss of non-safety related instrument air, valves MU3, CC1495, SW1356, SW1357, SW1358, SW1424, SW1429, and SW1434 would not have been capable of performing their intended safety functions for all required conditions during past plant operation. Therefore, LER 2003-001 was supplemented to include the valves determined by the AOV Reliability Program to be deficient. A description of each of these valves is given separately below.

The Safety Features Actuation System (SFAS) [JE] at the DBNPS is designed to automatically prevent or limit fission product and energy release from the core, to isolate the containment vessel [NH] and to initiate the operation of the engineered safety features equipment in the event of a loss-of-coolant accident or fuel handling accident inside containment. The SFAS also initiates protective actions in the event of a Main Steam line break. SFAS protective actions include signalling various valves to stroke to their safety position or sending a confirmatory signal for valves to remain in their safety position.

The Component Cooling Water (CCW) System [CC] circulates water through a closed cooling loop to provide cooling water to safety-related equipment and reactor auxiliary equipment within the Auxiliary [NF] and Containment [NH] Buildings, transferring the heat through the CCW heat exchangers [CC-HX] to the ultimate heat sink [BS] (Lake Erie) via the Service Water (SW) System [BI]. The system pressure is set by the CCW Surge Tank [CC-TK] located upstream of the CCW Pumps [CC-P]. During a Design Basis Accident, CCW supply to non-essential components is isolated, and cooling water is supplied only to essential components, including the Emergency Diesel Generator Jacket Cooling Water Heat Exchangers [LB-HX], Decay Heat Pumps [BP-P] and Bearing Housing Coolers [BP-P-CLR], High Pressure Injection Pumps' Bearing Coolers [BQ-P-CLR], the Decay Heat Removal Heat Exchangers [BP-HX], and Containment Gas Analyzer Heat Exchangers.

During normal plant operation, the Service Water (SW) System supplies SW to the CCW heat exchangers, Containment Air Coolers (CACs), Turbine Plant Cooling Water heat exchangers, and Emergency Core Cooling System (ECCS) room coolers for cooling. Operation of the ECCS room coolers is not required during normal plant operation for heat removal, however, SW flow through the room coolers is required to minimize corrosion of the cooling coils. During emergency operation, automatic valve sequencing aligns a redundant supply path of SW to safety-related components which include the CCW heat exchangers and CACs. A Safety Features Actuation Signal (SFAS) will cause SW pumps to start and realign valves to supply SW to essential equipment and isolate SW flow to non-essential equipment.

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

DESCRIPTION OF OCCURRENCE (continued):

Valve CC1495, CCW Auxiliary Building Non-Essential Header Isolation Valve [CC-ISV]

Valve CC1495 is a 16-inch butterfly valve which is normally open to provide cooling water to non-essential components such as the Spent Fuel Pool Heat Exchangers [DA-HX], Reactor Coolant Pump Seal Return Coolers, Reactor Coolant System Sample Cooler, Post-Accident Sample Coolers, Pressurizer Quench Tank Cooler, and various Clean Liquid, Miscellaneous Liquid, and Gaseous Radwaste System [WE] components. On an SFAS Level 3 signal, this valve closes to isolate non-essential loads and ensure that cooling water is available for Engineering Safety Features components and the Makeup pumps' gear and lube oil coolers. This valve also closes on low level in the CCW Surge Tank to ensure sufficient CCW flow is available to essential equipment. However, calculations showed that the safety-grade air accumulator installed at the valve to provide a source of motive power in the event of a loss of non-safety related instrument air is undersized. The available air volume was not sufficient to ensure the valve would fully close in the event of a loss of instrument air on an SFAS Level 3 signal or a low CCW Surge Tank level.

Valves SW1356, SW1357, and SW 1358, Service Water (SW) Return Valves from Containment Air Coolers (CAC) [BK-CLR]

These are 8-inch ball valves which are normally open on operating CACs to provide a flow path of SW from the operating CACs to the SW return header outside containment. The CACs provide a containment atmosphere cooling function during both normal operation and accident conditions. During normal operation, these valves provide temperature control for their respective operating CAC. In the event of a design basis accident, an SFAS Level 2 signal will automatically place the two operating CAC fans in low speed. Fan operation in low speed initiates an electrical signal to stroke the control valves in the SW return lines to full open to allow maximum SW flow through the operating CACs. In the event of loss of instrument air, an actuator spring provides the motive force to open the valves. However, these valves have a dual safety function.

For both normal operation and accident conditions, two of three CACs are in operation. The third CAC is placed in standby. This spare CAC will have its SW outlet valve closed during both normal and emergency operation based on containment isolation considerations. These valves are located outside containment. The valves are held in the closed position by air pressure. Since the design basis accident analysis considers a 30-day accident duration, the SW return valve on the spare CAC must retain its containment isolation function (remain closed) for 30 days. With a loss of instrument air, the air accumulators for these valves did not have sufficient capacity to meet this 30-day closure requirement.

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DESCRIPTION OF OCCURRENCE (continued):

Valves SW1424, SW1429, and SW1434, CCW Heat Exchanger SW Outlet Valves

These are 12-inch ball valves located on the SW return piping from the CCW heat exchangers. During normal operation, they serve as temperature control valves by throttling SW flow. During emergency operation, the outlet control valves for both the operating and standby CCW heat exchangers go to their full open position upon receipt of an SFAS Level 2 signal in order to maximize SW flow through the CCW heat exchangers. The valves are also to fail open upon loss of instrument air. These valves have a double-acting with spring assist cylinder actuator which requires the presence of air to stroke to its safety position. These valves were not provided with an air accumulator to assist in valve stroking. Based upon the results of a dynamic differential pressure test performed on SW1434, this spring force, alone, was determined to be inadequate to stroke the valves to their full open position.

Valve MU3, Reactor Coolant System (RCS) Letdown Isolation Valve [CB-ISV]

This 2.5-inch gate valve is normally open to allow letdown flow to pass from the Letdown Coolers [CB-CLR] to the purification demineralizers [CB-DM]. A small portion of reactor coolant is letdown for purification, chemical control, and degasification purposes during normal plant operation. This isolation valve closes on an SFAS Level 2 signal to isolate letdown flow as well as to provide for (outside) containment isolation. This valve is also designed to fail closed upon a loss of instrument air. An internal actuator spring will take this valve to its failed position. Calculations performed showed that the isolation valve would not fully close against a maximum Reactor Coolant System pressure of 2500 psig.

APPARENT CAUSE OF OCCURRENCE:

Lessons learned from the nuclear power industry's motor-operated and air-operated valve programs indicate that AOV performance can be enhanced by improvements in valve and actuator sizing, setting, testing, and maintenance. It was found that during the original procurement cycle, many AOV actuators were undersized. This was a result of vendors being provided with inaccurate system conditions in combination with less than conservative sizing methodology used at the time, and a lack of formal calculations supporting the design basis and appropriate settings for AOV actuators. There was also the practice of sizing AOV actuators with minimum built-in margin. Similar analytical deficiencies resulted in the design of the air accumulators, used to provide a source of motive power in the event of a loss of non-safety related instrument air, not being sufficient to ensure the valves would perform their intended safety function under all design conditions. This apparent cause applies to valve CC1495.

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APPARENT CAUSE OF OCCURRENCE (continued):

Valves SW1424, SW1429, and SW1434 were originally designed with air accumulators to assist in valve stroking. Plant modification 87-1315 replaced the valve assemblies in 1991 which included removal of the accumulators. These valves have a double-acting with spring assist actuator. The modification that installed these actuators stated that these valves are taken to their safety position by a mechanical spring. It was therefore concluded that the valves' accumulators were not needed and were removed. The apparent cause of this condition was an incorrect engineering assumption based on deficient design information.

As noted in the Description of Occurrence, valves SW1356, SW1357, and SW1358 have a safety function to provide containment isolation for non-operating CAC(s) following a LOCA. The accumulators for these valves were determined to have insufficient air capacity to keep the valves closed for a 30-day period, consistent with accident analysis assumptions. The apparent cause of this condition was an apparent lack of understanding at the time of plant's construction of the design and licensing basis and to correlate this information into the design of the accumulators for these valves and lack of design evaluation for the sizing of the accumulators.

As noted in the Description of Occurrence, the actuator for MU3 was designed with an internal actuator spring to stroke the valve to its fail closed position upon loss of instrument air. During normal plant operation when this valve is open, the spring is in a state of compression. Over time, this led to the degraded condition of "spring relaxation," a reduction in spring length caused by creep of the spring material under load. This spring relaxation reduced the free length of the spring which resulted in a degraded capability to close MU3 against the force of its higher range of design operating pressure. This spring relaxation phenomenon, which is caused by normal equipment operation over time, was not recognized for purposes of preventive maintenance.

ANALYSIS OF OCCURRENCE:

At the time of discovery there were no applicable Technical Specification operability requirements for the affected systems with the reactor defueled. However, the plant operated in this condition when the CCW System was required to be operable per TS 3.7.3.1, the SW System was required to be operable per TS 3.7.4.1, and Containment Isolation Valves were required to be operable per TS 3.6.3.1. Therefore, this issue represents a condition prohibited by the Technical Specifications, and is reportable in accordance with 10CFR50.73(a)(2)(i)(B).

While the reactor coolant Letdown Outlet Isolation Valve (MU3) would not have closed against a maximum RCS pressure of 2500 psig, a preliminary engineering evaluation determined that the valve would have closed against a differential pressure of 2030 psi. If a leak in the RCS caused pressure to decrease to the SFAS Level 2 actuation setpoint of 1600 psig, MU3 would have closed to perform its safety function of isolating containment and reactor coolant Letdown flow.

ANALYSIS OF OCCURRENCE (continued):

Additionally, valve MU2A is located inside of containment in compliance with General Design Criterion 55 to isolate letdown flow upon receipt of the same SFAS Level 2 signal. This motor-operated valve remained fully capable of isolating letdown flow at full RCS pressure. Therefore, containment isolation capability of Letdown flow was degraded but maintained.

The CCW Auxiliary Building Non-Essential Header Isolation Valve CC1495 would not have fully closed in the event of a loss of non-safety related instrument air in conjunction with either an SFAS Level 3 signal or a low CCW Surge Tank level. This latter event would likely be mitigated by automatic actions. In the event of a leak in non-essential equipment, a low level in the CCW Surge Tank would signal motor-operated valves CC5096 and CC5097, which are upstream of CC1495, to close. These two valves would then isolate the non-essential header regardless of the capability CC1495 to fully close in the event of a leak in non-essential equipment. In the event of an SFAS Level 3 signal in conjunction with a loss of non-safety related instrument air, the valve's safety grade air accumulator would not have had sufficient capacity to fully close the valve. However, motor-operated valves CC5096 and CC5097, noted above, are available to provide remote manual isolation of the non-essential CCW header.

As discussed in Description of Occurrence, the respective SW return valves from the operating CACs, SW1356, SW1357, and SW1358, would not have been capable of maintaining containment isolation capability consistent with accident analysis assumptions. However, this SW piping is a closed fluid system inside containment (not connected to the Reactor Coolant Pressure Boundary nor connected directly to the containment atmosphere). Therefore, failure to maintain the containment isolation function would not create a release pathway for post-accident radioactive material.

As discussed in the Description of Occurrence, the SW return valves from the CCW heat exchangers, SW1424, SW1429, and SW1434, would not have been capable of stroking to their full open position upon receipt of an SFAS Level 2 signal. Based on the results of a dynamic differential pressure test, it was concluded that the valves would only be able to stroke about 28 degrees from the closed position. Air accumulators which served to assist the safety function of the valves were removed in 1991 on the assumption that the actuator internal spring was sufficient to stroke the valves full open. The vendor information provided at the time of this modification suggested that the spring was strong enough to provide adequate torque. This assumption was incorrect and resulted from a lack of design bases for the valves and lack of calculations to support valve orientation, set-up, and sizing for risk significant AOVs.

LER 2003-001-00 had additionally identified valves CC1467 and CC1469, Decay Heat heat exchanger isolation valves[BP-HX-ISV], as potentially incapable of performing their intended safety function. Engineering calculation C-ME-016.04-035, Rev. 0, determined that the actuators for these two valves had positive

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ANALYSIS OF OCCURRENCE (continued):

margin for proper valve operation. Therefore, these valves would have been capable of performing their intended safety function under all postulated plant conditions.

Engineering calculation C-NSA-099.16-080 Revision 0, was developed to determine the increase in Core Damage Frequency (CDF), the Core Damage Probability (CDP), the increase in Large Early Release Frequency (LERF), and the Large Early Release Probability (LERP) for the conditions described in LER 2003-001. The results of this calculation determined that the increase in CDF associated with this event was 4.57E-7 / year. The increase in LERF associated with this event was 2.18E-11 / year. Based on the total operating time the condition was determined to have existed, the increase in CDP was 4.85E-6 and the increase in LERP was 2.31E-10. Therefore, these valve conditions were considered to have minimal safety significance.

CORRECTIVE ACTIONS:

An AOV Reliability Program is being implemented, in part, to ensure that AOV actuator sizing and setpoints are reviewed to verify and document their adequacy. As previously committed in DBNPS LER 2002-004, for Category 1 and 2 AOVs and their associated components, the design basis requirements, including the correct installed orientation, have been established in accordance with the AOV Reliability Program Manual. The requisite engineering documents have been developed to implement changes for category 1 and 2 AOVs. Modifications needed to restore these components to their design requirements have been completed with the exception of SW1356 which will be completed prior to restart. Post-modification testing has been performed to verify compliance with design bases for the completed modifications. SW1356 will be tested prior to restart. Identification of the AOV conditions presented in this LER are the direct result of implementation of the AOV Reliability Program which is intended to preclude recurrence of these conditions in the future. In addition, the 1991 modification to valves SW1424, SW1429, and SW1434 removed the air accumulators for these valves based upon an incorrect design/licensing assumption. The recently instituted Design Interface Evaluation process is intended to preclude future occurrence of similar conditions that can result from inadequate communications among organizations affected by a plant modification.

For valves CC1495, SW1424, SW1429, and SW1434, modifications have been implemented to install properly sized safety-related air accumulators. For valves SW1356, SW1357, and SW1358, modifications are being implemented to install properly sized safety-related high pressure nitrogen bottles. For valve

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

CORRECTIVE ACTION (continued):

MU3, a modification has been implemented to install a new valve actuator (which does not utilize spring force) and an additional safety-related air accumulator capable of closing the valve against maximum RCS differential pressure.

FAILURE DATA:

DBNPS LER 2002-004 documents similar problems discovered with air-operated valves for which the actuators did not have the capability to properly position the associated valve for all postulated conditions. It was during the performance of the corrective actions as stated in LER 2002-004 that the problems associated with the eight valves described above were discovered as an extent of condition.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

NP-33-03-001-02	PCAQR 97-1082	CR 99-2111	CR 02-07750
	CR 02-07781	CR 03-00830	CR 03-01040
	CR 03-01253	CR 03-02475	CR 03-04158
	CR 03-04878	CR 03-05628	CR 03-07859