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Dresden Generating Station
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ComEd

June 11, 1996

JSPLTR #96-0089

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

Licensee Event Report 96-004, Docket 50-249 is attached and is being submitted pursuant to 10CFR50.73(a)(2)(iv), which requires reporting of any event that results in unplanned manual or automatic actuation of any engineered safety feature, including the Reactor Protection System.

This correspondence contains the following commitments:

1. Vibration monitoring of the 3B Feedwater Regulating Valves (FWRV) will be performed to identify the flow level at which the greatest frequency levels are reached and to assure the frequency determined in the stress calculation is not maintained. (249-180-96-00401)

2. After vibration data is obtained, it will be determined if the 3B FWRV operational limitation should be revised. (249-180-96-00402)

3. The 3B FWRV will be modified during D3R14 to have Control Components Inc. (CCI) trim installed. (249-180-96-00403)

4. An industry review of the CCI trim will be performed to verify its operational performance. (249-180-96-00404)

5. A white paper analyzing this event will be prepared by the Station Manager, relative to conservative decision making concerning operating Unit 3 with only one FWRV open. This paper will be submitted to all ComEd Station Managers and INPO. (249-180-96-00405)

6. A 50.59 Safety Evaluation will be performed on the April 27, 1996, isolation of the 3A FWRV. (249-180-96-00406)

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7. The potential contributing causes will be reviewed. If any additional contributing causes are identified, they will be reported in a subsequent supplement. (249-180-96-00407)

8. The Unit 2 12HGA17S63 type relays will be tested and replaced as required. (249-180-96-00408)

9. Establish a trending process of the Electronic Work Control Process component data. (249-180-96-00409)

10. Determine if the relay problem is a 10CFR21 reportable failure and report as appropriate. (249-180-9600410)

11. A review of the PCIS reset philosophy will be performed. Procedure improvements will be implemented as appropriate. (249-180-96-00411)

If you have any questions, please contact Pete Holland, Dresden Regulatory Assurance Supervisor at (815) 942-2920 extension, 2714.

Sincerely,

J. Stephen Perry
Site Vice President
Dresden Station

Enclosure

cc: H. Miller, Regional Administrator, Region III
NRC Resident Inspector's Office
Illinois Department of Nuclear Safety

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 3	05000249	1 OF 11

TITLE (4)
Reactor Vessel Level Transient Resulting In Reactor Scram and Emergency Core Cooling System Actuation Caused By Feedwater Regulating Valve Stem Separation

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
									None	
05	15	96	96	-- 004 --	00	06	11	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N

POWER LEVEL (10): 83

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)
	20.2203(a)(1)		20.2203(a)(3)(ii)	X	50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)	X	OTHER
	20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)		50.73(a)(2)(vii		
	20.2203(a)(2)(v)		50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Paul K. Garrett, Plant Engineering Ext. 2713	(815) 942-2920

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
B	BD	62	G080	Yes					
E	SJ	FCV	C635	Yes					

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 May 15, 1996, at approximately 1043, with Unit 3 in the run mode at approximately 83% power, a feedwater transient was experienced which resulted in the initiation of an automatic reactor Scram due to low reactor water level and Emergency Core Cooling System (ECCS) actuation on low low reactor water level. The plug from the 3B Feedwater Regulating Valve (FWRV) separated from the stem resulting in an isolation of feedwater flow to the reactor. The cause of the stem separation was fatigue cracking. In addition, 2 Group I isolation valves improperly re-opened upon resetting the isolation signal. The cause for isolation valves re-opening is a manufacturing defect and is potentially reportable under 10CFR21. Corrective actions include; replacing the 3B valve stem to existing specifications, operational limits and vibration monitoring of the valve upon resumption of power operations, and future reconfiguration of the 3B FWRV, and replacing the relays. The overall safety significance was determined to be minimal because all safety systems performed as required and there was no danger to health and safety at any time.

EVENT IDENTIFICATION:

Reactor Vessel Level Transient Resulting In Reactor Scram and Emergency Core Cooling System Actuation Caused By Feedwater Regulating Valve (FWRV)[SJ] Stem Separation

A. PLANT CONDITIONS PRIOR TO EVENT:

On May 15, 1996, at 1043 Dresden Unit 3 was in the "Run" Mode at 976 psig reactor pressure. The reactor was operating at approximately 83 percent core thermal power and no deliberate changes were being made to reactor power. Unit three had been on line since November 7, 1995. All emergency equipment was available and normal offsite electrical power was available. Normal routine surveillances were in progress on the Containment Cooling Service Water System. 3A and 3C Reactor Feed Pumps [SJ] were operating. The 3A FWRV was closed due to a body to bonnet leak since April 27, 1996, and it's inlet valve was shut. The 3A Low Flow FWRV was closed. The 3B FWRV was controlling reactor water level in automatic and was maintaining normal reactor water level at approximately +30 inches.

Dresden Unit 2 and 3 share a common control room. Dresden Unit 2 was in the "Run" Mode at approximately 55% reactor power.

B. DESCRIPTION OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv), any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF) including the Reactor Protection System (RPS).

On May 15, 1996, at approximately 1043, with Unit 3 in the run mode at approximately 83% core thermal power, a feedwater transient was experienced on Unit 3 which resulted in the initiation of an automatic reactor scram, Group I, II, III isolation signals [JM], Unit 3 and Unit 2/3 Diesel Generators [EK], and Emergency Core Cooling System (ECCS) [JE] actuation on low low reactor water level.

The initial indication of a problem were alarms [IB] received in the Control Room [NA]. These alarms were the "Feedwater Reg Station high vibration" and "3B FWRV actuator trouble" alarms. Feedwater flow indications showed a quick drop in feedwater flow from 8 millon pounds (lbs) per hour to zero lbs flow in less than 3 seconds. The Unit 3 Nuclear Station Operator (NSO) (Licensed Reactor Operator) immediately acknowledged the alarms and verified that the 3B FWRV positioning indication showed 100% feedwater flow demand, which should have meant that the 3B FWRV was open and providing adequate feedwater to the reactor. The Unit 3 NSO also identified that the Reactor Feed Pump (RFP) and Total Feedwater Flow recorder indicated there was no flow through the feedwater system. This indication meant that the reactor was not getting feedwater flow adequate to maintain water level at the existing steam flow out of the reactor vessel.

Four seconds after the preceding feedwater alarms were received, reactor level had lowered to 25 (+25) inches above instrument zero (instrument zero is 12 feet above the top of active fuel), as indicated by the Reactor Low Level alarm. The Control Room received an automatic reactor scram at eight seconds into the event on low water level (at approximately +17 inches reactor water level), concurrent with the Group II and III Containment Isolation signals. All Control Rods (CRD) [AA] inserted as designed. Containment isolation actuation occurred and all valves closed as designed. Containment isolation was verified as complete.

The Unit 3 Supervisor (Licensed Senior Reactor Operator) took command of the event response from the time the reactor scram was received. The Unit 2 Supervisor (Licensed Senior Reactor Operator) ordered the termination of all Unit 2 surveillances, obtained a replacement for his duties on Unit 2, and proceeded to Unit 3 to assume the duties of the Shift Technical Advisor (STA). The Center Desk NSO (Licensed Reactor Operator) entered Unit 3 and placed himself at the 903-3 panel to perform any ECCS manipulations which may be required. The Unit 3 NSO (Licensed Reactor Operator) took the immediate actions of Dresden General Procedure (DGP) 2-3, Unit 2(3) Reactor Scram, depressing the manual scram pushbuttons, placing the reactor mode switch in shutdown and performed verification of full control rod insertion.

One minute and 10 seconds into the event, the reactor level reached -53.9 inches resulting in an automatic start of the Unit 3 and Unit 2/3 Diesel Generators, initiation of the High Pressure Coolant Injection (HPCI) [BJ] system, and a Group I isolation. This Group I isolation removed the Main Condenser [BS] as a primary heat sink.

The Center Desk NSO announced the initiation of HPCI, and while monitoring HPCI system performance, initiated torus cooling. HPCI injection began within 15 seconds of the initiating signal and reached full flow of approximately six thousand gallons per minute within 22 seconds. At one minute and 29 seconds into the event the reactor water level then reached -59 inches below instrument zero (7 feet above the top of active fuel) and reactor water level began to increase. This was the lowest reactor water level reached during the event.

The Main Turbine/Generator [TA, TB] then tripped on reverse power and auxiliary power transferred successfully. The Unit 2 Auxiliary NSO performed verification of these events by performing Dresden Operating Abnormal (DOA) procedure 5600-01, "Turbine Trip", and DOA 6000-01, "Generator Trip", with no deficiencies noted. Manual control of HPCI was ordered approximately 3 minutes into the event for level control, resulting in reactor level steady at 30", within the level band of +8 to +48 inches as directed by the Unit 3 Supervisor. The appropriate emergency operating procedures were entered and executed correctly.

At 1050, the Unit 3 NSO secured the 3C RFP as a result of the abnormally high pressure indications in the Condensate System [SD], which he had been monitoring since the onset of the event. At 1051 the 3A RFP was secured.

At 1057, the HPCI system was allowed to automatically trip on high reactor water level (+48 inches), due to the addition of water from the CRD system. As HPCI coasted down, its turning gear did not automatically engage as the turbine came to a stop. The Center Desk NSO took the appropriate actions to initiate turning gear operation. Additionally, the NSO entered DOP 2300-04 for securing of the HPCI system with an initiation signal present. Subsequently it was identified that the HPCI Auxiliary Coolant Pump was running hot. The Aux Coolant pump was then secured.

At 1101, a report that 3 of 4 Turbine Building sump pumps were running continuously was received in the Control Room. Equipment Attendants (Non-Licensed Operators) were dispatched to the field to look for possible feedwater/condensate system leaks. It was identified that the high level in the sumps was the result of the water relieving from the Condensate Booster Pump suction relief valves. After ensuring that appropriate reactor inventory control was in place, the condensate and feedwater systems were secured. The Unit 3 Supervisor then directed the Center Desk NSO to take manual control of the Isolation Condenser [BL] to control reactor pressure, establishing a controlled plant cooldown. Radiation Protection was notified of the Isolation Condenser initiation and dispatched personnel to rope off the area outside the Isolation Condenser vent to prevent access.

At 1111, with no indication of high radiation levels in the main steam lines (MSL), the Control Room attempted to reset the Group I isolation signal to re-establish the Main Condenser as a heat sink. Approximately 3 minutes after the Group I isolation was reset, it was observed that the 1A Main Steam Isolation Valve (MSIV) and the 3-220-45 Recirculation Sample Valve had re-opened without Operator intervention. Normal reset would have resulted in the valves maintaining their shut position. These valves were then closed from the Control Room. No further attempt was made to open the MSIVs and the Main Condenser remained unavailable as a heat sink. An Unusual Event was declared at 1122 per procedure and conservative operating practice. The proper notifications were made and manning of the Technical Support Center (TSC) was not required.

The NRC Emergency Operations Center was notified via ENS of the event at 1137 (CDT) on May 15, 1996, pursuant to 10CFR50.72(b)(1) and 10CFR50.72(b)(2)(ii).

All safety systems functioned as expected and designed, and Plant conditions were stabilized. At 1200, an investigation team was formed by the Station Manager and the investigation into the event commenced. At 2316 (CDT), an ENS notification was made to terminate the Unusual Event.

3B FWRV Modification

The 3B FWRV was originally supplied as a double ported, air operated Copes-Vulcan series D-100.

The 3B valve internals and actuator were modified in 1988 (M12-3-87-45B) because of a feedwater transient experienced on Unit 3 (LER 87-013, docket number 050-249). These modifications included replacing the double-seat plug assembly with a cage and plug assembly. The pneumatic actuator air supply was modified to allow for control dampening. The valve's pneumatic actuator was later replaced in 1991 with a hydraulic actuator by modification M12-3-88-57.

The 3B FWRV stem was installed in accordance with an Original Equipment Manufacturer (OEM) recommended upgrade procedure (WR92246) with maintenance complete on March 30, 1992. This upgrade was a result of an inspection that was completed on December 1, 1991, that identified cracking in the valve plug. The valve inspection was in response to an Institute of Nuclear Plant Operation's (INPO) Nuclear Network OE (3263) that communicated an event at another utility where a fracture on the stem of a similar type of Copes-Vulcan valve resulted in a complete shear of the stem and failure of the valve. It was determined at the time that the follow up inspections at Dresden would be performed every second refueling outage. The next inspection of the 3B FWRV was scheduled for next refueling outage D3R14 (September 1996). In addition, upgraded Control Components Inc. (CCI) trim was scheduled for installation during D3R14.

From the time the modification was installed on 3B FWRV in 1992 until failure, the valve saw approximately 32 months of actual operation.

3B FWRV Operation

During normal operation at full power, the 3A FWRV is in automatic and the 3B FWRV is approximately 25% open in manual. However, on April 27, 1996, with the reactor at approximately 87% power, it was decided to take the 3A FWRV closed, due to valve body to bonnet leakage, and the 3B FWRV was placed into automatic providing the full feedwater flow required by the reactor. The Low Flow FWRV was closed and in manual. Unit 3 remained in this condition.

On April 28, 1996, about 10 hours after the 3A FWRV was closed, the 3B FWRV was determined to be capable of supporting Unit 3 operation, at approximately 40% open, with sufficient margin to facilitate transient conditions. A review of the FSAR section 10.4.7 was performed by the Unit Supervisor and System Manager at that time. The FSAR indicated that potential flow problems could occur at greater than 700 MWe with the operation of a single FWRV passing 100% of reactor rated flow. Although Unit 3 was in operation at 712 MWe in coastdown, it was determined that the 3B FWRV operation was satisfactory based on no excessive vibration as noted by field walkdowns and reactor feed flow was less than 100% of reactor rated flow.

The plan was to temporarily fix the 3A FWRV during the next scheduled load drop, tentatively scheduled the beginning of June. Senior Station management were aware of and agreed with this decision. However, a formal review of the potential single failure of the valve and the resultant loss of normal feedwater flow was not performed.

3B FWRV Stem Separation

The valve stem separated at a location 1/16 of inch below flush with the valve plug and approximately 3/4 of an inch above the initial stem thread. The stem failed due to high cycle, low stress fatigue cracking. Beach marks were observed radiating out from two crack initiation sites. The crack arrest fronts covered approximately 95% of the fracture face and converged at the area of final fracture. The multiple crack arrest fronts and the small area of final fracture imply high cycle fatigue, under bending loads, at a low nominal stress.

Review of the previous work packages identified no errors or omissions that could affect the stress levels in the valve stem except that the run out values for the stem and plug assembly were not obtained. The stem run out had been checked and was satisfactory.

During re-assembly of the stem to plug connection, an inspection of the run out of the stem found that the plug and stem were properly aligned axially. However, a radial off-set of 0.006 of an inch was noted which was slightly outside the OEM's recommendation of 0.004 of an inch. It was also found that the line of final failure was essentially perpendicular to the axis of maximum off-set. This indicates that the valve was probably oscillating in the direction of this minor imbalance. This also added a small amount to the bending moment, but is considered insignificant to the flow induced vibration.

The stresses affecting the plug/stem interface are most likely a result of the plug moving within the cage in a translation and angular motion. This results in a bending moment created at the junction, which with sufficient number of applications, caused the stem to fail.

Isolation Valves Improperly Re-Open

The HGA relay which provides the seal in logic and light indication for the Primary Containment Isolation System (PCIS), experienced mechanical binding/ friction between the contact finger pivot point and the relay coil bracket. Failure of this relay does not impair the isolation valve function if a valid Group I isolation signal is initiated.

The procedure used during the resetting of the Group I isolation provides a statement that the Isolation Valves' control switch be placed to "close" prior to resetting the Group I isolation. However, this information was contained in a procedural note which are not to contain action statements and thus require no action. If the valve's control switch would have been placed to the closed position, the 3-203-1A Main Steam Isolation Valve (MSIV) and the 3-220-45 Recirculation Sample isolation valve would not have opened during the Group I isolation signal reset.

HPCI Turning Gear/Cooling Pump

It was determined that the procedure which controls securing HPCI did not provide sufficient direction. This resulted in the Aux Coolant pump running without an adequate flow path and manual initiation of the HPCI turning gear.

C. CAUSE OF EVENT:

The cause of the event is a design/manufacturing defect, NRC cause code B, of the 3B FWRV.

3B FWRV Stem Separation

The 3B FWRV's stem separated from the plug resulting in an isolation of the feedwater flow. The failure mode for the valve stem is fatigue. The root cause of the stem separation is the 3B FWRV was not evaluated by the vendor, during the upgrade process, for the stress cycles it experiences from flow induced vibration. Contributing cause was management decision making which should have been more conservative in that operating the feedwater system with only one FWRV placed the system into a condition where it was susceptible to a single failure and isolation of the normal feed flow. Operating with the 3A FWRV isolated was predicated upon continued reliable operation of the 3B FWRV. This contributed to reaching the reactor level scram setpoint and did cause reaching the reactor level ECCS initiation setpoint.

Additional contributing causes include: ineffective corrective actions from the 2A FWRV stem failure investigation in 1987 (LER number 87-024, Docket number 050-0249), duration of the scheduled inspection of the 2B FWRV was too long and additional stresses from the run out of the shaft to the plug and the angular mis-alignment of the bonnet.

Isolation Valves Improperly Re-opening

Thirteen of the Unit 3's HGA relays were tested at ComEd's test facility. Seven of the relays exhibited stickiness to the point of test failure. Each of the failed relays were purchased from Nutherm (a dedication company) and were identified with a mold #1 marking in the phenolic contact finger carrier. The root cause for the contact of the finger and bracket portions of the HGA relay is attributed to the clearance and tolerance values in the manufacture/ assembly process. This relay problem is a potential 10CFR21 reportable failure.

Contributing was that the procedure used during the Group I isolation signal reset was inadequate. In addition, precursor occurrences of the relay failure were found to have occurred in 1994, but a sufficient component trending program was not in place to identify potential common mode failure.

HPCI Turning Gear/ Cooling Pump

The root cause for the HPCI Cooling pump heat up and manual initiation of the turning gear is a deficient procedure.

D. SAFETY ANALYSIS:

A review of selected safety systems and parameters was performed to verify proper response following this event. Plant systems operated as expected and followed the sequence of events as outlined in the FSAR.

The ECCS demand system was initiated within design and the HPCI system operated as described in the FSAR to reestablish normal reactor water level. The lowest indicated water level during the event was minus 59 inches which meant that approximately seven feet of water remained above the top of the active fuel region throughout the event.

The Group I isolation initiated as designed and the containment was isolated by the closure of all twelve isolation valves. Two valves did, however, reopen upon reset of the isolation signal by the operators. The re-opening of the 3-203-1A, Main Steam Isolation Valve (MSIV), and the 3-220-45, Recirculation Sample isolation valve, upon resetting the Group I isolation signal had minimal safety impact. There is a second redundant isolation valve in each of the respective lines and these redundant isolation valves remained closed when the Group I isolation signal was reset. If an additional valid Group I isolation signal would have recurred, the valves in question would have gone closed.

As a result of the Group I isolation experienced during the scram, the Isolation condenser was used for reactor decay heat removal and reactor pressure control as the plant was cooled down to a cold shutdown condition. Because of the single tube heat exchanger design of the Isolation condenser there exists the possibility of a monitored release through the Isolation Condenser exhaust if this condenser is used as the heat sink for the reactor decay heat. A water and steam mixture was released into the environment during the event per the design of the Isolation Condenser as it was used for decay heat removal to bring the reactor to a cold shutdown condition. Conservative calculations were performed which demonstrate that the radioactive release was well below the regulatory limits established in 10CFR50, Appendix I.

The overall safety significance was determined to be minimal because all safety systems performed as required and there was no danger to health and safety at any time.

E. CORRECTIVE ACTIONS:

Feedwater Flow

A liquid penetrant test was performed on the new stem for the 3B FWRV to verify that there were no relevant surface indications and the new stem was installed into the 3B FWRV. In addition, based on vendor recommendations, when the valve was re-assembled, the bonnet was verified to be square, and the stroke was verified to be smooth.

An ultrasonic inspection was performed on the 3A FWRV stem to verify that there were no sub-surface discontinuities.

A stress calculation was performed on the 3B FWRV to determine the induced stresses by the applied loads from the plug and guide to the stem. Based on this stress level, the number of cycles needed to fatigue the material to failure were determined from graphs of stress vs. number of cycles. Based on the number of cycles to failure and expected operational time plus 2 months (6 Months), the maximum allowable frequency was calculated. Based on follow up analysis of the valve resonant frequency, it was determined that the 3B FWRV could reliably operate for the remainder of the operating cycle. The remaining FWRVs at Dresden Units 2 and 3 are equipped with different internals which have different flow characteristic within the valve, and do not subject the valve to flow induced vibration.

Vibration monitoring of 3B FWRV will be performed to identify the flow level at which the greatest frequency levels are reached and to assure the frequency determined in the stress calculation is not maintained. (249-180-96-00401)

After vibration data is obtained, it will be determined if the 3B FWRV operational limitation should be revised. (249-180-96-00402)

The 3B FWRV has had an operational limit of no more than 25% open placed on it. This is to minimize valve stem stresses.

The 3B FWRV will be modified during D3R14 to have Control Components Inc. (CCI) trim installed. (249-180-96-00403)

An industry review of the CCI trim will be performed to verify its operational performance. (249-180-96-00404)

A contingency plan has been determined for various FWRV configurations.

An independent review was performed on the evaluation and the findings. The review concurred with identified causes and corrective actions.

A white paper analyzing this event will be prepared by the Station Manager, relative to conservative decision making concerning operating Unit 3 with only one FWRV open. This paper will be submitted to all ComEd Station Managers and INPO. (249-180-96-00405)

Walkdowns of the associated piping systems were performed. No significant problems or anomalies were identified.

A 50.59 Safety Evaluation will be performed on the April 27, 1996, isolation of the 3A FWRV. (249-180-96-00406)

Isolation Valves Improperly Re-opening

The relays were tested/ inspected to verify proper operation. New relays were installed on Unit 3 as required.

The potential contributing causes will be reviewed. If any additional contributing causes are identified, they will be reported in a subsequent supplement. (249-180-96-00407)

An operability determination was performed for Unit 2, document ID# 4975044. The conclusion of the determination was that the safety function of the Group I isolation is unaffected. However, Operator action will be required to prevent unexpected valve movement upon the reset of the Group I isolation signal until the Unit 2 relays have been tested.

Operations personnel were notified of the need to take the control switches for the affected valves to the closed position prior to resetting a Group I isolation.

The Unit 2 12HGA17S63 type relays will be tested and replaced as required. (249-180-96-00408)

Establish a trending process of the Electronic Work Control Process (EWCS) component data. (249-180-96-00409)

Determine if the relay problem is a 10CFR21 reportable failure and report as appropriate. (249-180-9600410)

A review of the PCIS reset philosophy will be performed. Procedure improvements will be implemented as appropriate. (249-180-96-00411)

HPCI Turning Gear/ Cooling Pump

DOP 2300-04 was revised to provide sufficient information concerning the operation of the Cooling Pump and the turning gear initiation.

F. PREVIOUS OCCURRENCES:

LER/Docket Number	Title
87-024/ 050-249	Unit 2 Reactor Scram On Low Level Due to 2A Feedwater Regulating Valve Failure

On August 21, 1987, a failure occurred on the 2A FWRV. The 2A was a Copes Vulcan D100 valve, with the original double ported internals and a different style stem than the 3B FWRV. The valve stem separated at approximately the same location as the 3B. At the time of failure the 2A was in automatic and the 2B was 25% open in manual, Unit 2 was at approximately 93% power and the valve was in service for approximately 30 months prior to failure. The root cause was attributed to fatigue and no further review was performed to identify the fatigue initiator. The corrective action was to install a new stem and plug in the 2A valve and weld the stem/ plug together. Corrective action to prevent recurrence was to review the modification planned for Unit 3 (from the 1987 Unit 3 feedwater transient, LER 87-013, docket number 050-0249) and determine if a similar modification for Unit 2 would be appropriate to reduce valve vibration and fatigue. The Hush trim was installed on the 3B valve in 1988 (M12-3-87-45B). Later modifications installed CCI Drag trim on the 2A, 2B, and 3A FWRVs. The CCI drag trim modification on the 3B FWRV was scheduled for the upcoming Unit 3 D3R14 refuel outage.

G. COMPONENT FAILURE DATA:

Manufacture	Nomenclature	Model/ Part Number
General Electric (dedicated by Nutherm)	Time Delay Relay	12HGA17S63
Copes-Vulcan	FWRV	Series D-100 with Hush trim