CP&L

Carolina Power & Light Company
P.O. Box 10429
Southport, NC 28461-0429

APR 16 1996

SERIAL: BSEP 96-0152
10 CFR 50.73

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NO. 50-325 AND 50-324/LICENSE NO. DPR-71 and DPR-62
LICENSEE EVENT REPORT 1-96-003

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. George Honma at (910) 457-2741.

Sincerely,

R. P. Lopriore-Plant General Manager
Brunswick Nuclear Plant

SFT/sft

Enclosures
1. Licensee Event Report
2. Summary of Commitments

cc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

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NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 1	05000324	1 OF 6

TITLE (4)

Dual Unit Shutdown Due To Service Water Pump Inoperability

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
									BSEP Unit 2	05000325
03	17	96	96	-- 03 --	00	04	16	96	FACILITY NAME	DOCKET NUMBER 05000

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

OPERATING MODE (9)								
		20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)		20.2203(a)(1)		20.2203(a)(3)(i)		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)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)		50.36(c)(2)	X	50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Tabor, Sr. Analyst, Regulatory Affairs	(910) 457-2178

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	BI	P	J105	Y					

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 March 9, 1996, at approximately 1800 hours, with Unit 1 operating at rated power and Unit 2 in shutdown to support the B212R1 scheduled refuel outage, the 2A Nuclear Service Water pump (SWP) tripped on overcurrent after operating for approximately 20 minutes. Initial troubleshooting indicated that the pump was binding and disassembly was required to determine the cause. On March 13, 1996, the investigation determined that the pump impeller thrust ring had become loose due to thrust ring retainer bolt failure which allowed the impeller to slip on the shaft and resulted in pump binding and the overcurrent condition. The bolts failed due to corrosion.

On March 14, 1996, an operability assessment was initiated to ascertain whether a common cause failure condition existed with the remaining SWPs. During this assessment similar bolt degradation was discovered on the 1 B Nuclear and 1 B Conventional SWPs. Based on these findings the remaining uninspected SWPs on both Units were declared inoperable on March 17,1996, at 1825 hours and reactor shutdown of both Units was initiated. Unit 2 reactor shutdown commenced at 1834 hours and Unit 1 reactor shutdown commenced at 2205 hours. An Unusual Event was declared at 1920 hours.

The investigation results indicate the primary cause of the bolt failures was corrosion induced by galvanic coupling of the retainer bolting and other pump components. Consequently, efforts were initiated to repair the 10 SWPs as necessary which included replacing the affected bolting with suitable material. Upon completion of the repairs necessary to return to service the number of SWPs required by the Technical Specifications for Unit operation, Unit 2 startup was initiated on March 21, 1996 and Unit 1 startup commenced on March 23, 1996.

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Brunswick Steam Electric Plant, Unit 1	05000324	96	-- 03 --	00	2 OF 6

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

TITLE

Dual Unit Shutdown Due To Service Water Pump Inoperability

INITIAL CONDITIONS

On March 9, 1996, Unit 1 was operating at rated power and Unit 2 was in shutdown to support the B212R1 scheduled refuel outage.

EVENT NARRATIVE

On March 9, 1996, at approximately 1800 hours, the 2A Nuclear service water pump (SWP) tripped on overcurrent after operating for approximately 20 minutes. An attempt to restart the pump was unsuccessful. Preliminary investigation determined that the pump was bound and disassembly was required to troubleshoot the condition. The Service Water System (BI) pumps (BI/P) are two stage vertical turbine pumps (Model # 27CC) manufactured by the Johnston Pump Company. Upon disassembly of the pump on March 13, 1996, the four 3/8" socket head cap screws which attach the upper impeller thrust ring retainer to the impeller were discovered corroded. The corrosion resulted in the failure of the screws which allowed the split thrust ring sections to become loose and the impeller to slip on the shaft. This condition resulted in the pump binding problem. Additionally, these inspections revealed corrosion of the eight 3/8" hex head bolts which hold the four shaft bearings in place. The thrust ring retainer bolts for the lower impeller assembly exhibited minimal indication of corrosive activity. Repair activities were initiated to restore pump operability.

On March 14, 1996, an operability assessment was initiated to obtain more data and ascertain whether a common cause failure condition existed with the remainder of the 10 SWPs. On March 15, 1996, the 1B Conventional SWP was removed from service and disassembled for inspection. Investigation of this SWP revealed that the bolting that attaches the retainer to the impeller exhibited signs of corrosive activity similar to the degradation observed on the 2A Nuclear SWP. Additionally, on March 17, 1996, after restoring the 2A Nuclear SWP to service, the 1 B Nuclear SWP was removed and disassembled for inspection. Similarly, this inspection revealed bolting degradation like that observed in previously inspected SWPs.

Based on the apparent generic bolt corrosion issue, the remaining uninspected SWPs on both Units were declared inoperable on March 17, 1996, at 1825 hours. In accordance with the requirements of the Technical Specifications, reactor shutdown of both Units was initiated. Unit 2 reactor shutdown commenced at 1834 hours and Unit 1 reactor shutdown commenced at 2205 hours. An Unusual Event was declared at 1920 hours. In accordance with the requirements of 10 CFR 50.72 (b)(1)(i), a one hour report was made at 1932 hours. On March 18, 1996, at 2133 hours, once both Units reached a cold shutdown condition, the Unusual Event was terminated.

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

Subsequent inspections of the remaining SWPs indicated that the bolting exhibited varying degrees of corrosion. The stages of degradation observed ranged from nearly intact bolts, to partially corroded but sound bolts, to severely corroded bolts where only a portion of the bolt shank was intact. Attachment 1 tabulates the observed as-found condition of the SWPs. The bolting external to the pump exhibited little or no corrosion.

Investigation into the cause of the component failures was initiated utilizing a fault tree approach and the assistance of CP&L metallurgical engineering personnel and several industry metallurgical/corrosion consultants and root cause specialists. The investigation results indicate the primary cause of the bolt failures was corrosion induced by a galvanic coupling of the retainer bolting and other pump components. The retainer bolting was B/SB 164 UNS N04400 (Monel 400) and the pump components consist of A/SA 351 Grade CF3M (type 316L SS), A276 Type 316A SS, and B/SB 688 UNS N08367 (AL6XN SS). Consequently, the SWPs were repaired as necessary. These repairs included replacing the thrust ring retainer cap screws and shaft bearing bolting with SB 574, alloy N10276 (Hastelloy C-276) material.

With this material change, the cathodic surface area of the galvanic couple will be smaller than the anodic surface area (Hastelloy is more cathodic than either 316 or AL6XN). Additionally, the Hastelloy C-276 material is proven to provide superior corrosion resistance in seawater environments. Follow-up inspections will be performed to assess the performance of the new bolting material.

On March 20, 1996, Carolina Power & Light issued INPO Operational Experience report #7747 to inform the industry of this event. Upon completion of the repairs necessary to return to service the number of SWPs required by the Technical Specifications for Unit operation, Unit 2 startup was initiated on March 21, 1996 and Unit 1 startup commenced on March 23, 1996.

This event is reportable in accordance with the requirements of 10 CFR 50.73 (a)(2)(i) and 10 CFR 50.73(a)(2)(vii) in that the failure of the SWP bolting resulted in a shutdown required by the Technical Specifications and represents an event where a single cause or condition caused two independent trains or channels to become inoperable in a single system designed to mitigate the consequences of an accident.

CAUSE OF EVENT

The SWP component failures resulted from a design change which designated what was determined to be an inappropriate material for a specific function. The material configuration of the SWP's was changed by plant modifications 82-220L and 82-221L. The material changes associated with these plant modifications were developed in 1990 and 1991 and the modifications implemented during 1993 and 1994. These modifications specified the use of Monel 400 bolting material.

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

Investigation results indicate that the appropriate engineering organization provided design inputs and reviews related to material selections. However, the material evaluations of the changes associated with the SWP plant modifications were not intrusive enough to determine the internal bolting would subsequently fail due to accelerated galvanic corrosion.

Due to the fact that the galvanic potential of the materials used in the SWPs is similar throughout the pump, the critical component for recognizing the possibility of an accelerated galvanic attack was the large cathode (316 or AL6XN) to anode (Monel 400) area ratio between the pump components and the internal bolting. Specific guidance was not in place during the 1990/1991 period which would require an examination of the cathode to anode area ratio.

CORRECTIVE ACTIONS

The SWPs were repaired as necessary including replacement of corroded bolting with Hastelloy material.

The following reviews of the Service Water System were performed:

> Material changes associated with configuration and design changes to the Service Water System since 1991, which includes at least 2 cycles of operation for each unit, were reviewed to ensure proper material selections were made. Additionally, material changes associated with completed maintenance work orders on the Service Water System from 1991 through March of 1996 were reviewed to ensure proper material selections were made. Considerations for these reviews included the following material degradation modes: galvanic corrosion (includes cathode to anode area ratio review), pitting/crevice corrosion, erosion, wear, cavitation, and radiation effects.
>
> Material condition visual inspections were performed on accessible portions of the Service Water System.
>
> No conditions affecting equipment operability were identified during these reviews. The other minor conditions that were identified during these reviews are being addressed in accordance with the Corrective Action and Corrective Maintenance Program.

In addition to the above reviews, an assessment of the Service Water System will be performed by August 30, 1996, to determine if component material selection and application were adequate.

Service Water System configuration and design changes which involve material changes and are currently developed and approved for installation will be reviewed by May 31, 1996, to ensure proper material selections were made.

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

A procedure will be developed by June 30, 1996, to provide detailed guidance for selecting materials associated with design changes. Prior to the development of the procedure, real time training will be completed by April 30,1996, to discuss the event and establish guidance for proper selection of materials associated with configuration and design changes.

The design specification process will be evaluated by June 30, 1996, to determine the need for additional enhancements.

SAFETY ASSESSMENT

One Nuclear and four Conventional SWPs were determined to be operable based on visual examination and testing of the existing bolting. Additionally four of the remaining five SWPs (Two Nuclear and two Conventional) were considered to be functional (e.g., capable of starting and supplying design flow) based on recent data.

To meet minimum functional design basis requirements, one Nuclear SWP is required to automatically start and supply necessary safety related cooling water flow during the first ten minutes following a design basis accident (DBA) with or without a loss of offsite power (LOOP). This pump will supply four Emergency Diesel Generator (EDG) SW requirements, which are the safety related SW cooling loads during this time period. After the first ten minutes following a DBA, one additional SWP on the opposite unit (Nuclear or Conventional) is required to supply safety related cooling loads. These loads would be aligned with each pump supplying its own Unit's EDGs, Residual Heat Removal heat exchangers, and Vital Header cooling loads.

Following a DBA with a LOOP, one Nuclear SWP would be required to avoid a Station Blackout (SBO). At the time of discovery one operable Nuclear SWP and two functional Nuclear SWPs were available. The simultaneous loss of these three SWPs is not credible. In addition, Abnormal Operating Procedure 18.0, Nuclear Service Water System Failure, could be entered to supply the additional necessary cooling water flow by manually starting one of the four operable Conventional SWPs on the Nuclear Service Water header. These pumps can be manually started from the main control room.

Therefore, there is reasonable assurance that the Service Water System was functionally capable of satisfying design basis requirements prior to March 17, 1996, when Units 1 and 2 were shutdown.

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

PREVIOUS SIMILAR EVENTS

A previous event involving material selection was reported in LER 1-95-019. This event involved a component material change resulting from an equivalency evaluation. This evaluation did not include the necessary design inputs from the appropriate engineering organization. The corrective actions associated with this similar event focused on non-modification engineering products to determine whether proper material selections were made.

In this event, the component material changes were developed in accordance with the plant modification process and appropriate engineering organization design inputs were provided. Therefore, this product was not subject to the review and corrective actions from LER 1-95-019.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Essential Service Water	BI
Pump	P

ATTACHMENT 1

SERVICE WATER PUMPS - AS FOUND CONDITION

PUMP	AS FOUND CONDITION	OPERABLE	MATERIAL CONFIGURATION INSTALLATION DATE
1A Nuc SWP	Upper thrust ring retainer fasteners moderately corroded with portion of fastener heads in tact, retainer ring had not dropped down, impeller in proper location	Yes	10/94
1B Nuc SWP	Upper thrust ring retainer fasteners corroded away, retainer ring dropped down, impeller dropped down and sustained some damage which was not significant enough to degrade pump performance	No	6/94
1A Conv SWP	Upper thrust ring retainer fasteners moderately corroded with portion of fastener heads in tact, retainer ring had not dropped down, impeller in proper location	Yes	7/94
1B Conv SWP	Upper thrust ring retainer fasteners corroded away, retainer ring dropped down, impeller dropped down and sustained some damage which was not significant enough to degrade pump performance	No	8/93
1C Conv SWP	Upper thrust ring retainer fasteners significantly corroded with portion of fastener heads in tact, retainer ring had not dropped down, impeller in proper location	Yes	9/94
2A Nuc SWP	Upper thrust ring retainer fasteners corroded away, retainer ring dropped down, impeller dropped down and sustained some damage which was not significant enough to degrade pump performance	No	10/93
2B Nuc SWP	Upper thrust ring retainer fasteners corroded away, retainer ring in place but one thrust ring found in strainer, impeller had not dropped down and was not damaged.	No	3/94
2A Conv SWP	Upper thrust ring retainer fasteners moderately corroded with portion of fastener heads in tact, retainer ring had not dropped down, impeller in proper location	Yes	12/93
2B Conv SWP	Upper thrust ring retainer fasteners corroded away, retainer ring had not dropped down, impeller had not dropped down and was not damaged	No	2/94
2C Conv SWP	Upper thrust ring retainer fasteners had moderate corrosion with full fastener heads in place, retainer ring in place, impeller in place and not damaged	Yes	11/94

Enclosure
List of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
Real time training will be conducted to discuss this event and establish guidance for proper selection of materials associated with configuration and design changes.	4/30/96
Service Water System configuration and design changes which involve material changes and are currently developed and approved for installation will ve reviewed to ensure proper material selections were made.	5/31/96
A procedure will be developed to provide detailed guidance for selecting materials associated with design changes.	6/30/96
The design specification process will be evaluated to determine the need for additional enhancements.	6/30/96
An assessment of the Service Water System will be performed to determine if component material selection and application were adequate.	8/30/96
A follow-up inspection will be performed to assess the performance of the new bolting material.	3/31/97

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

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 INFORMATION AND 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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley Nuclear Plant - Unit 1	0 5 0 0 0 3 4 8	1 OF 0 3

TITLE (4)

Technical Specifications Action Statement Requirement Not Met For Solid State Protection System Testing

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
0 3	2 2	9 6	9 6	- 0 0 1 -	0 0	0 4	1 6	9 6	J M Farley - Unit 2	0 5 0 0 0 3 6 4
									FACILITY NAME	

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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

20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	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 Specify in Abstract below or in NRC Form 366A
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)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
R. D. Hill, General Manager - Nuclear Plant	3 3 4 8 9 9 - 5 1 5 6

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)

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-space typewritten lines) (16)

At 0900, on March 22, 1996, with Unit 1 and 2 in Mode 1 operating at 100% power, it was discovered that Farley Nuclear Plant (FNP) had been periodically operating in a condition that was prohibited by Technical Specifications (TS). TS 3.3.2 Action Statement 17 associated with Containment Purge and Exhaust Isolation states, "with less than the minimum channels operable, operation may continue provided the containment purge and exhaust valves are maintained closed." However, the Containment Mini Purge and Exhaust System was not always being isolated during Solid State Protection System (SSPS) [JE] testing.

This event was caused by cognitive personnel error due to a misinterpretation of TS 3.3.2 Action Statement 17. Operations personnel have been provided instructions to secure Containment Mini Purge and Exhaust during appropriate SSPS testing. Applicable test procedures have been revised to ensure the Containment Mini Purge and Exhaust System is isolated during appropriate SSPS testing.

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

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 INFORMATION AND 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.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
Joseph M. Farley Nuclear Plant - Unit 1	05000348	96	- 001	- 00	02 OF 03

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

Plant and System Identification

Westinghouse -- Pressurized Water Reactor
Energy Industry Identification System codes are identified in the text as [XX].

Description of Event

At 0900, on March 22, 1996, with Unit 1 and 2 in Mode 1 operating at 100% power, it was discovered that Farley Nuclear Plant (FNP) had been periodically operating in a condition that was prohibited by Technical Specifications (TS). TS 3.3.2 Action Statement 17 associated with Containment Purge and Exhaust Isolation states, "with less than the minimum channels operable, operation may continue provided the containment purge and exhaust valves are maintained closed." However, the Containment Mini Purge and Exhaust System was not always being isolated during Solid State Protection System (SSPS) [JE] testing.

Cause of Event

Failure to isolate the Containment Mini Purge and Exhaust System during SSPS testing was due to cognitive personnel error. This was due to a misinterpretation of TS 3.3.2 Action Statement 17. Specifically, FNP SSPS generates a Containment Purge and Exhaust Isolation signal from any of the following: automatic or manual Safety Injection (SI), manual Phase "A" Containment Isolation, or manual Containment Spray/Phase "B" Actuation. TS 3.3.2 Action Statement 13 allows one channel to be bypassed for up to four hours for testing for automatic actuation logic of SI, Containment Spray and Containment Isolation Phase "A" and Phase "B." However, this four hours is not provided in Action Statement 17 for an automatic Containment Purge and Exhaust Isolation, which is also part of Containment Isolation. As a result of Containment Purge and Exhaust Isolation being a part of Containment Isolation, Action Statement 17 has been misinterpreted to isolate the Containment Mini Purge and Exhaust System only if one channel was inoperable for reasons other than for testing or if it was discovered inoperable during testing.

Safety Assessment

This event is reportable because of the failure to isolate the Containment Mini Purge and Exhaust valves to meet the requirement of Technical Specification 3.3.2 Action Statement 17 for Containment Purge and Exhaust which states, "with less than the minimum channels operable, plant operation may continue provided the containment purge and exhaust valves are closed."

This event did not prevent the fulfillment of the safety function of the Containment Purge and Exhaust System because the system would receive an isolation signal from the redundant SSPS train or from redundant radiation monitors which are independent of SSPS. In addition, FNP Emergency Response Procedures direct the operators to verify that Containment Purge and Exhaust has isolated early in any event requiring purge isolation.

Corrective Action

Operations personnel have been provided instructions to secure Containment Purge and Exhaust during appropriate SSPS testing.

Applicable test procedures have been revised to ensure that the Containment Purge and Exhaust System is isolated during appropriate Solid State Protection System testing.

Additional Information

The following LERs involved TS action statements not being met due to misinterpretation of Technical Specifications:

LER 95-002: Missed Surveillance for Inoperable Axial Flux Difference Monitor Alarm.