CATEGORY 1

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FACIL:50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400

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SUBJECT: LER 95-011-01:on 951105,reactor trip/safety injection during SSPS testing due to failure of a relay contact & unplanned ESF actuation.Caused by component/equipment failure. Block relay replaced & testing completed.W/960322 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:Application for permit renewal filed. 05000400

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SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 95-011-01

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed supplement to Licensee Event Report 95-011 is submitted. This supplement provides additional information pertaining to the failure mode of the relay contacts that failed to remain closed and caused the unplanned Reactor Trip/Safety Injection event on November 5, 1995.

Sincerely,

J. W. Donahue
General Manager
Harris Plant

MV

Enclosure

c: Mr. S. D. Ebneter (NRC - RII)
Mr. N. B. Le (NRC - PM/NRR)
Mr. D. J. Roberts (NRC - HNP)

IE22 1/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
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: 50.0 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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Reactor Trip/Safety Injection during Solid State Protection System testing due to the failure of a relay contact, and unplanned ESF actuation during troubleshooting following the Reactor Trip/SI.

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
11	05	95	95	-- 011 --	01	03	29	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100 %

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

	20.2201(b)		20.2203(a)(2)(v)		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)	X	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)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Michael Verrilli, Sr. Analyst - Licensing	(919) 362-2303

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	JE	RLY CNTR	P297	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 November 5, 1995, A-Train Engineered Safety Feature Actuation System (ESFAS) slave relay surveillance testing was in progress. While performing the test portion that verifies the operability of the Main Steam Line Isolation Signal circuitry, a "Low Steam Line Pressure Reactor Trip/SI" signal was generated. This occurred when a contact failed to maintain continuity on the K809 relay and eliminated the "block" function to the "A" Main Steam Isolation Valve (MSIV), which resulted in the "A" MSIV closing. As the valve closed and turbine throttle valve position remained constant, increased steam load was carried by "B" and "C" Steam Generators (SGs). This resulted in a pressure decrease in the "B" and "C" main steam lines due to increased steam flow. The rate compensation feature associated with the low steam line pressure SI initiated a low steam line pressure Reactor Trip/SI Signal for "C" steam line. Automatic systems responded as required and main control room operators took appropriate actions to stabilize the plant in Mode-3 (Hot Standby). An Unusual Event was declared at 0805 due to the ECCS actuation. The UE was then exited at 0912 based on the termination of SI.

On November 6, 1995, continued ESFAS slave relay testing was being performed following the Reactor Trip/SI event from the previous day. During this testing the Auxiliary Feedwater (AFW) Flow Control Valves fully opened from their original throttled position. While this valve actuation was in accordance with system design, it was not recognized in the procedure. Thus, operators performing the test did not expect the valve operation. The test being performed is normally performed in Mode-1 with the AFW FCV's fully open. With the plant in Mode-3, the valves are throttled to control Steam Generator (SG) level. The operators recognized that the valves were opening and reestablished Steam Generator level control prior to exceeding the normal operating band. During the investigation of this actuation, a similar occurrence was identified where the AFW FCV's partially opened during testing on October 30, 1994. At that time the actuation was not recognized as reportable, but should have been.

This LER revision provides additional information from laboratory failure analysis of the K809 relay contacts which failed to remain closed and caused the unplanned Reactor Trip/Safety Injection event on November 5, 1995.

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

EVENT DESCRIPTION:

On November 5, 1995, with the plant operating in Mode-1 at 100% power, A-Train Engineered Safety Feature Actuation System (ESFAS, EIIS Code: JE) slave relay surveillance testing (OST-1044) was in progress. While performing section 7.3 of the test procedure, the portion of the test that verifies the operability of the Main Steam Line Isolation Signal circuitry, a "Low Steam Line Pressure Reactor Trip/Safety Injection (SI)" signal was generated. This occurred at 0737 hours, when a contact failed to maintain continuity on the K809 relay and eliminated the "block" function to the "A" Main Steam Isolation Valve (MSIV, EIIS Code: SB), which resulted in the "A" MSIV closing. As the valve closed and turbine throttle valve position remained constant, steam load was carried by "B" and "C" Steam Generators (S/Gs). This resulted in a pressure decrease in the "B" and "C" main steam lines due to increased steam flow. The rate compensation feature associated with the low steam line pressure SI initiated a low steam line pressure Reactor Trip/SI Signal for "C" steam line. Automatic safety equipment functioned as required except for the "closed" indication on one valve, SP-941, (Hydrogen Monitor isolation valve, EIIS Code: TK). It was later determined that the valve closed as required but experienced a position indication proximity switch problem.

Due to the SI, the Reactor Coolant System (RCS, EIIS Code: AB) was being filled to solid plant conditions by the injection flow. The main control room operators proceeded through the Emergency Operating Procedure (EOP) network as required to secure SI. Although progress through the EOP procedure flow paths was timely, it did not prevent the plant from going solid. The solid plant conditions resulted in an increase in RCS pressure, which lifted a pressurizer PORV. A liquid/steam mixture was released by the pressurizer PORV to the Pressurizer Relief Tank (PRT). During this time the PORV actually cycled approximately 58 times. This resulted in an overpressure condition in the PRT and one of the two rupture disks ruptured as required to limit pressure in the tank. Approximately 1200 gallons of water from the PRT spilled over into the containment sump through the rupture disk. The control room staff took actions necessary to stabilize the plant and establish operation in Mode-3 (Hot Standby). An Unusual Event (UE) was declared at 0805 due to the ECCS actuation. The UE was then exited at 0912 based on termination of SI and stabilization of the plant in Mode-3. Operations personnel performing the test at the time of the Reactor Trip/SI signal took prompt and appropriate actions to preserve existing plant and system conditions, which were critical in identifying the deficient K809 relay contacts.

On November 6, 1995, continued ESFAS Slave Relay testing was being performed following the Reactor Trip/SI event from the previous day to verify the operability of other SSPS "block function" relays. At 1509 hours, while performing section 7.3 of the A-Train, K635 Relay SI Block Circuit Test (OST-1044), the Auxiliary Feedwater (AFW, EIIS Code: BA) Flow Control Valves (FCV's) fully opened from their original throttled position. While this valve actuation was in accordance with system design, it was not recognized in the procedure. Thus, operators performing the test did not expect the valve motion. Based on the guidance of NUREG-1022, the opening of the AFW FCV's was considered an unplanned ESF actuation. OST-1044 is a quarterly interval surveillance test that is normally performed in Mode-1 with the AFW FCV's fully open. Due to the fact that these valves have been open during previous testing, no valve motion occurred. With the plant in Mode-3 following the Reactor Trip/SI, the valves were throttled to control Steam Generator (SG) level. The operators recognized that the valves were opening and reestablished Steam Generator level control prior to exceeding the normal operating band. During the investigation of this actuation, a similar occurrence was identified where the AFW FCV's partially opened during testing on October 30, 1994. At that time the actuation was not recognized as reportable, but should have been. Corrective actions for this occurrence involved a revision to the K640 Relay "Go Circuit" portion of the test procedure, but the "block circuit" portion of the test was not recognized, nor revised as needed.

CAUSE:

The cause of the Reactor Trip/SI was component/equipment failure. One set of contacts on the K809 "blocking" relay failed to maintain continuity as required when the block signal was generated. Additional analysis will be performed to determine the cause and failure mode of the relay contacts. The Unplanned ESF Actuation was caused by an inadequate procedure. Section 7.3 of OST-1044 did not provide proper guidance to ensure that operators performing the test were aware that the AFW Flow Control Valves would receive an open signal while testing the K635 block circuit. The October 30, 1994 AFW FCV actuation was also caused by a deficient procedure and was not reported due to a misunderstanding of reporting requirements on the part of Operations personnel that were involved with testing at the time.

Additional investigation was performed by Potter & Brumfield Products Division to determine the failure mechanism for the K809 relay contact failure. This investigation revealed that the low current load from the SSPS test circuit did not have enough arc energy to burn through the surface tarnish, which is naturally present on silver contacts of this age. Corrective actions have been taken to procedurally address this conditon.

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TEXT CONTINUATION

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

SAFETY SIGNIFICANCE:
There were no significant safety consequences as a result of either event. The reactor trip and safety injection actuations were initiated by plant conditions resulting from an equipment failure. Safety systems responded as required to ensure plant safety and operators appropriately responded to verify system response and stabilize the plant in Mode-3. The Unplanned ESF Actuation was also a result of plant systems responding as designed. Though the opening of the AFW Flow Control Valves was unexpected by operators performing the testing, S/G level was never outside of the normal operating band.

PREVIOUS SIMILAR LERs:
No similar LERs have been reported.

CORRECTIVE ACTIONS COMPLETED:
1. The K809 Block Relay was replaced and A-Train ESFAS Slave Relay Testing was completed.
2. An engineering analysis, including a walkdown inspection was completed for the secondary plant to evaluate possible water hammer effects.
3. An inspection was performed in the containment building for the areas in the vicinity of the PRT that contain environmentally qualified equipment.
4. An inspection was performed to assess the condition of the PRT Rupture disks. This included a "foreign material exclusion" inspection for rupture disk fragments.
5. An evaluation was performed to assess the effects of the water (approximately 1,200 gallons) that spilled into the containment sump from the PRT rupture disk.
6. An inspection and evaluation was performed to assess the condition of the Pressurizer PORV that lifted and cycled during the event.

Note: For each of the above inspections and/or evaluations (C/A's 2-6) no discrepancies were identified that would adversely effect or preclude plant startup.

7. Surveillance procedures OST-1044 and OST-1045 were revised to provide operators guidance pertaining to the operation of AFW Flow Control Valves during K635 Relay testing.
8. Reporting requirements related to unplanned ESF actuations were reemphasized with Operations personnel.
9. Additional review was completed to ensure that similar AFW System testing deficiencies do not exist in other procedures.
10. Additional investigation into the cause and failure mechanism for the K809 relay contacts was performed. The results are provided in this supplement.
11. Testing procedures OST-1044 and OST-1045 were revised to require cycling the test switches several times between the "NORMAL" and "PUSH TO TEST" positions prior to depressing the test switch. This will help ensure that test switch contacts are wiped sufficiently to remove any tarnish prior to opening of the slave relay contact.

EIIS CODES:
Engineered Safety Feature Actuation System, EIIS System Code: JE Component Code: RLY CNTR
Main Steam System, EIIS System Code: SB Component Code:ISV
Hydrogen Monitor Isolation Valve, EIIS System Code: TK Component Code: ISV
Reactor Coolant System, EIIS System Code: AB
Auxiliary Feedwater System, EIIS System Code: BA Component Code: FCV