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SUBJECT: LER 89-023-00:on 891223,installation of low leakage core causing PR NIS to indicate non-conservatively.
W/8 ltr.

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NOTES:Application for permit renewal filed. 05000400

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R. B. RICHEY
Manager
Harris Nuclear Project

FEB - 5 1990

Letter Number: HO-900029 (O)

U.S. Nuclear Regulatory Commission
ATTN: NRC Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 89-023-00

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report is submitted seven days beyond the 30 day requirement. This was discussed with NRC personnel in a telephone conference on January 22. The report is in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

R. B. Richey

R. B. Richey, Manager
Harris Nuclear Project

RBR:dgr

Enclosure

cc: Mr. R. A. Becker (NRR)
Mr. S. D. Ebneter (NRC - RII)
Mr. J. E. Tedrow (NRC - SHNPP)

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MEM/LER-89-23/1/OS1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): SHEARON HARRIS NUCLEAR POWER PLANT - UNIT 1

DOCKET NUMBER (2): 0 5 0 0 0 4 0 0

PAGE (3): 1 OF 0 8

TITLE (4): Installation of Low Leakage Core Causing Power Range Nuclear Instrumentation to Indicate Non-Conservatively

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 NAMES	DOCKET NUMBER(S)
1 2	2 3	8 9	8 9	0 2 3	0 0	0 2	0 5	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 4 2

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

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	X 50.73(a)(2)(vii)	OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME: Andrew J. Howe - Senior Specialist

TELEPHONE NUMBER: AREA CODE 9 1 9 | 3 6 2 - 2 7 1 9

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 fifteen single-space typewritten lines)* (16)

On December 23, 1989, at 0230, following a refueling outage, the initial heat balance surveillance test on the reactor core showed a significant difference between actual and indicated reactor power level for the Power Range Nuclear Instruments (PR NIS). Actual power was 41.5% while indicated power on the PR NIS was 28.1%, a 48% error. The gain was immediately adjusted on the PR NIS.

The cause of the mismatch was the installation of a low leakage core loading pattern with no compensating adjustments made to the PR NIS to account for reduced neutron flux at the detector. No formal program to identify cycle specific requirements of power ascension following refueling outages existed. Procedures which would have made necessary adjustments to the PR NIS were not performed due to procedural deficiencies and personnel error, and because plant personnel and management were not aware of the significant impact of the low leakage core on the PR NIS. In addition, the mismatch between the PR NIS and other indications of power level (turbine load, core delta temperatures) was not detected in a timely manner.

A formal power ascension program will be implemented prior to the next refueling startup. The plant startup procedure will be revised to include monitoring of diverse power level indications and resolution of any discrepancies. Additional training of personnel and management is being conducted and appropriate changes to training programs will be implemented. A Human Performance Evaluation of the event is being conducted, as well as an internal investigation by personnel not associated with the plant.

Description:

On December 18, 1989, the plant was preparing for startup following a refueling outage in which a low leakage core loading pattern was installed. The Power Range Nuclear Instruments (PR NIS) reactor trip setpoints were adjusted to 50% of rated thermal power for initial criticality, instead of the normal Technical Specification value of 109%, per the direction of the plant Operations Manager.

Technical Specification 3.3.1 requires the PR NIS to be operable for Mode 1, power operation, and Mode 2, startup. However, calibrations of the PR NIS sensors are not required to be completed until sufficiently high power levels are reached to provide accurate data on actual core power levels. Specification 4.0.4 requires all surveillance testing to be completed within specified intervals prior to entry into a mode or condition where the component is required to be operable. An exception to this requirement is provided in Specification 3.3.1 for the PR NIS daily sensor calibrations using a secondary heat balance. The trip setpoints for the PR NIS bistables had been adjusted using the detector currents which existed during the previous cycle until adjustments could be made based on actual cycle three testing.

Mode 2 entry occurred on December 20 at 0323, and initial criticality occurred at 0947. Physics testing commenced, and the PR NIS trip setpoints were adjusted down to 25% of rated thermal power per Special Test Exception 3.10.3 of Technical Specifications. Upon completion of this testing, a decision was made not to return the setpoints to 50%, but instead to adjust them as per Technical Specification requirements. Due to an inoperable Steam Generator Safety Relief Valve, Technical Specification 3.7.1.1 limited the maximum PR NIS high flux trip setpoint to 87%, so the PR NIS setpoints were established at 85% instead of the normal value of 109%.

On December 22, at 0516, the main generator was synchronized to the grid and power escalation towards 30% power commenced. Due to secondary chemistry parameters, power escalation was not continued, and power was stabilized at approximately 28% indicated power per the PR NIS.

Since power levels were stable, a heat balance to determine actual reactor power was performed. This test is required daily per Surveillance Requirement 4.3.1.1, Table 4.3-1, item 2.a, when the reactor power level is above 15%. The heat balance was performed per plant procedure OST-1004 on December 23 at 0230, and the results of this test showed a 48% error, with actual power at 41.5% with the lowest indicated PR NIS at 28.1%. Per Technical Specifications, the PR NIS gain was adjusted to reflect actual power level.

On December 30, further reviews of this situation were conducted at the request of the NRC resident inspector to determine the safety implications of the PR NIS being adjusted non-conservatively on initial startup. It was not realized by the involved plant personnel, or by their management, that the discrepancy between actual and indicated nuclear power at low power levels would result in more significant deviations at full power, which ultimately affect the actual power level at which the high neutron flux reactor trip would occur. With the high flux trip set at 85% due to the steam generator safety valve being inoperable, and with the existing mismatch ratio applied, the high flux reactor trip would have occurred at a nominal 125.5% power. This exceeds the 109% limit in Technical Specifications and the 118% limit assumed in the Final Safety Analysis Report.

Cause:

The root cause of this event is the lack of formal control over power ascension activities for startups following refueling outages, and insufficient management attention to the planning and implementation of power ascension. Information regarding changes in neutron flux leakage for the cycle three core and the significant impact on PR NIS indications was available in the Cycle Three Nuclear Design and Operations Package. Plant procedures which would have made the compensating adjustments were not performed because of procedure deficiencies and personnel error, and because plant personnel and management responsible for these activities were not aware of the magnitude of the impact of the new core loading pattern on the PR NIS high flux reactor trip setpoint.

With no compensating adjustments to the PR NIS made prior to startup, the mismatch between indicated nuclear power and other indicators of power level, such as turbine load and core delta temperature, should have been detected and investigated. This did not occur.

There are a number of contributing factors which resulted in the failure to make compensating adjustments to the PR NIS prior to startup, and the failure to detect the mismatch between the PR NIS and other indications after startup and power escalation.

1. Procedure EPT-008, "Intermediate Range Detector Setpoint Verification," was to be performed because the intermediate and source range detectors had been physically relocated further from the core. (This relocation was done to reduce the indicated neutron flux level of the source range detectors. This would allow reactor criticality to be achieved with an adequate margin to the neutron flux level on the source range detectors at which a reactor trip occurs, and would also allow criticality to be achieved prior to the point at which the operator normally de-energizes the source range detectors.) A

reactor engineer decided not to perform EPT-008 because the adjustments which had been made on the intermediate range instrumentation, prior to startup, were determined to be within the required accuracy, and because EPT-008 did not specifically cover physical relocation of detectors. Involved personnel and their management were not aware of the need to perform EPT-008 for the PR NIS compensation for the low leakage core in addition to the required adjustment of the intermediate range instrumentation to compensate for their physical relocation.

2. EPT-008 was revised to provide for readjustment of PR NIS for low leakage core loadings, but the title of the procedure was not revised to reflect the increased scope of the procedure. Personnel who were involved in this revision process were no longer associated with the reactor engineering group at the time of the startup of cycle three.

3. Extremely cold weather added an additional burden to operations personnel to perform duties related to monitoring and protecting plant equipment from the low temperatures during the initial power operation of cycle three.

4. The cold weather, combined with improvements to the cooling tower made during the outage, caused the operators to expect a higher generator output.

5. The power demands present, due to the record cold weather, led management to delay scheduled optional testing at the 30% power level, and go directly to 50% power. The power ascension testing is not a requirement of Technical Specifications or the Operating License, so delaying the testing seemed to be reasonable and prudent.

6. Attention of reactor engineers and operations personnel was directed to the intermediate range instrumentation response. This was due to conditions experienced during previous startups, in which the permissive to block the intermediate range high flux reactor trip was reached at a flux level very close to the level at which the intermediate range high flux trip would occur.

7. Reactor engineers had worked a significant amount of time to support control rod inspections and the receipt of new control rods, which diverted their attention from preparing for power ascension testing. In addition, there had been changes to the reporting structure and supervision of the reactor engineering group.

8. Operator training for cycle three operations was conducted several months prior to the actual startup of cycle three.

Safety Significance:

An evaluation of the plant conditions existing at the time of discovery of the mismatch was conducted by the Nuclear Fuels Section.

The following reactor trips receive input from the PR NIS:

1. Overtemperature Delta Temperature. The difference in neutron flux levels between the top and bottom of the core is used to compensate the trip setpoint for power imbalances. This input is provided for accidents where the flux shape is skewed to the extreme top or bottom of the core, such as control rod withdrawals or ejections. The power increase for the control rod ejection event is extremely high, so that the analysis is relatively insensitive to the actual setpoint of the trip. Protection for the control rod withdrawal is discussed in item six below.

2. Overpower Delta Temperature. While this trip circuitry includes the capability to compensate for events with skewed flux shapes as is done for the Overtemperature Delta Temperature trip, the delta flux input is zeroed out per plant design. As a result, there is no impact on the Overpower Delta Temperature channel operability.

3. Power Range High Positive Flux Rate. This trip is normally set at 5% power increase with a 2 second time constant. The trip setpoint could have been as high as 7.5%. The analysis of the Rod Control Cluster Assembly (RCCA) Ejection accident takes credit for this trip. However, due to the extremely high rate of increase in core power predicted to occur for the design basis RCCA Ejection accident, the analysis is relatively insensitive to the actual setpoint of the rate trip. Thus with the rate trip occurring at approximately 7.5% rather than 5% power increase over two seconds, there is no appreciable impact on the consequences of this accident.

4. Power Range High Negative Flux Rate. Similar to the positive rate trip, the normal setpoint is 5% power decrease with a 2 second time constant, which could have been as high as 7.5%. The analysis for the RCCA Misalignment accident takes credit for this trip. The consequences of this design basis accident are greatest at 100% power level. Since the maximum power level achieved during the event was 42%, the impact of the error in this trip setpoint would not be significant.

5. Power Range Neutron Flux - Low Setpoint. This trip is normally set at 25% power, so the trip setpoint could have been as high as 37.5%. Three events take credit for this trip: Excessive Heat Removal Due to Feedwater System Malfunctions - Zero Power, Uncontrolled Rod Control Cluster Assembly (RCCA) Bank Withdrawal from Subcritical, and RCCA Ejection. For the Excessive Heat

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

Removal accident, the results are bounded by the Uncontrolled RCCA Bank Withdrawal. For that accident, and for the RCCA Ejection, the flux rise is extremely rapid so that the error in the trip setpoint would not affect the consequences of the accidents.

6. Power Range Neutron Flux - High Setpoint. This trip was set at 85% power, so the trip setpoint could have been as high as 125.5%. Four accident analyses take credit for this trip: Uncontrolled RCCA Bank Withdrawal from Subcritical, RCCA Ejection, Startup of Inactive Reactor Coolant Loop, and Uncontrolled RCCA Bank Withdrawal from Power.

   - For the RCCA Bank Withdrawal - Subcritical and the RCCA Ejection, the flux rise is extremely rapid so that the consequences are insensitive to the trip setpoint.

   - For the inactive loop startup, the P-8 (single loop loss of flow) interlock setpoint provides protection since the core power level increases when the inactive loop is started to a power level above P-8 before the loop flow reaches a value sufficient to clear the low flow trip setpoint. The P-8 interlock is assumed to be set at 79% in the FSAR. However, Technical Specifications require it be set at 49%, so applying the PR NIS error brings the P-8 setpoint to 75%, which is less than the assumed 79%. Thus, this accident remains bounded.

   - For the RCCA Bank Withdrawal from Power, a reanalysis of this accident scenario for the 10% power case and a 42% power case was conducted. The PR NIS high flux trip setpoint was analyzed at 135%. This analysis showed that the acceptance limits for the transient were still met with the non-conservative trip setpoint.

Based on this information, it is concluded that the plant never operated in a manner that departed from the design basis or significantly challenged plant safety.

The plant could have operated at higher power levels if the secondary chemistry parameters had been within limits. The plan was to increase power to approximately 50% prior to the conduct of further power ascension testing. Once the plant reached approximately 60% actual power (approximately 41% indicated), turbine runbacks would occur since only one feedwater train would be operating. Problems were experienced when start attempts were made on the "1A" Main Feedwater Pump on December 21, 1989, (refer to Licensee Event Report 89-19). Management direction was to not attempt a start on this pump until after the Turbine Mechanical Overspeed Trip testing scheduled for the following week. This test was to be conducted at zero turbine load. Therefore, 60% actual power level is considered the highest credible power the plant could have operated.

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If the safety valve inoperability had not occurred, the high flux setpoint would probably have been set to its normal Technical Specification limit of 109%.

Under these conditions, an analysis shows that the accident consequences would have remained bounded by the existing safety analyses, by operation of the Overpower Delta Temperature reactor trip on a RCCA Bank Withdrawal, which was the only accident analysis for which the non-conservative PR NIS trip setpoint could not be demonstrated to provide protection.

No previous similar events have been reported.

This event is reportable per 10CFR50.73(a)(2)(vii) as a single cause (low leakage core installation) which resulted in multiple inoperable channels in the Reactor Protection System.

Corrective Actions:

1. All required calibrations of the PR NIS detectors have been completed for cycle three operations.

2. A formal program to identify the cycle specific requirements of the Power Ascension Test Program is being developed and will be in place prior to the next refueling startup.

3. Procedure EPT-008 is being revised to eliminate the conflicts in its title and scope and incorporate other improvements identified during the investigation of this event. Other procedures used for power ascension are being reviewed for similar deficiencies, and will be corrected prior to next use.

4. The plant startup procedure will be changed to require comparison of PR NIS to other diverse indications of reactor power, and resolution of any gross discrepancies.

5. Training on this event and its safety significance is being given to Operations and Reactor Engineering personnel, as well as to members of the plant supervisory staff. Additional training of selected technical staff and management personnel is planned, as well as changes, where appropriate, to licensed operator training and requalification training.

6. Practices for scheduling cycle-specific training will be reviewed and revised as appropriate.

7. A Human Performance Evaluation of the event is being conducted.

8. An internal investigation is being conducted by personnel not associated with the plant.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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<u>EIIS Codes</u>:

Reactor Core	AC
Nuclear Instrumentation	IG
Steam Generator Safety Relief	SB:RV