UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 25, 2000

EA-00-097

Carolina Power & Light Company
ATTN: Mr. James Scarola
Vice President - Harris Plant
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, NC 27562-0165

SUBJECT: NRC INTEGRATED INSPECTION REPORT 50-400/00-01

Dear Mr. Scarola:

On April 1**,** 2000, the Nuclear Regulatory Commission (NRC) completed an inspection at your Shearon Harris facility. The enclosed report presents the results of that inspection. The results of that inspection were discussed with you and other members of your staff on April 6, 2000.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.
The NRC identified three issues of low safety significance that have been entered into your corrective action program, and are discussed in the summary of findings and in the body of the enclosed inspection report. The three issues were determined to involve violations of NRC requirements, but because of their low safety significance the violations are not cited. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the NRC Resident Inspector at Shearon Harris Nuclear Power Plant; and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

/RA/

Brian Bonser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket No.: 50-400
License No.: NPF-63

Enclosure: Inspection Report

c w/encl: (See page 3)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	50-400/00-01
Licensee:	Carolina Power & Light (CP&L)
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	February 20, 2000 - April 1, 2000
Inspectors:	J. Brady, Senior Resident Inspector R. Hagar, Resident Inspector E. Testa, Senior Radiation Specialist (Section 20S2)
Approved by:	B. Bonser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

Shearon Harris Nuclear Power Plant, Unit 1
NRC Inspection Report 50-400/00-01

The report covers a six-week period of resident inspection. In addition, it includes the results of an announced inspection by a regional radiation specialist.

The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED) and was determined by the Significance Determination Process in draft Inspection Manual Chapter 0609 (see attachment 1).

Cornerstone: Mitigating Systems

- GREEN. A non-cited violation of the maintenance rule (10 CFR 50.65) was identified for failing to set goals and monitor the performance of the steam dump system when functional failures of the low-low reactor coolant system average temperature (Tave) interlock (P-12) to steam dump valves 1MS-109 and 1MS-110 occurred on December 14, 1999. The functional failures showed that the licensee had not demonstrated the performance of the system was being effectively controlled through the performance of appropriate preventive maintenance as required by 10 CFR 50.65(a)(2). Permissive P-12 shuts the steam dump valves when reactor coolant system Tave decreases to 553°F after a reactor trip. The safety significance was low because the main steam isolation valves could be closed if the steam dumps failed to operate. (Section 1R12)

- GREEN. A non-cited violation of the fire protection program was identified for failing to maintain adequate procedures for the installation of fire barriers where penetration seals interface with electrical raceway fire barrier wrap. This resulted in six examples where safe shutdown cables did not meet fire protection program requirements. The safety significance was low because for each example identified a diverse safe shutdown function was provided in another fire area to accomplish the same function. Consequently safe shutdown could have been achieved with the available equipment. (Section 1R05)

Cornerstone: Barrier Integrity

- GREEN. A non-cited violation of 10CFR50, Appendix B, Criterion XVI, “Corrective Action” was issued for failure to take corrective actions to stop activities which periodically rendered inoperable both trains of the Control Room Emergency Filtration System. Because the violation did not involve an actual job dose, and because the violation did not involve more than two occurrences in an 18-month period, the occupational radiation safety significance determination process determined that the safety significance was low. (Section 1R22.2)

Report Details

The unit operated at 100% of rated thermal power for the entire period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R03 Emergent Work

a. Inspection Scope

The inspectors reviewed the following emergent items, as described in the referenced Work Requests/Job Orders (WR/JOs) and/or Action Requests (ARs), to verify that the licensee had taken the necessary steps to demonstrate that emergent work activities were adequately planned and controlled to avoid initiating events, and to verify that the licensee ensured the functional capability of accident mitigation systems:

Reference	Title/Description
00-ABFI1	Shunt resistor problems with turbine driven AFW isolation valve 1AF-137
00-ABFJ1	Shunt resistor problems with turbine driven AFW steam supply valve 1MS-70
00-ABDS1	Failure of the “A” train undervoltage relay for the “A” reactor coolant pump

b. Observations and Findings

No findings were identified and documented through this inspection.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors reviewed plant documents and observed equipment to verify that the motor driven auxiliary feedwater (AFW) pumps were correctly aligned during a turbine driven AFW pump outage.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed current ARs, work orders, and impairments associated with the plant fire protection features. The inspectors reviewed the status of ongoing surveillance activities to determine whether they were current to support the operability of the fire protection systems. In addition, the inspectors observed the fire protection barriers, fire detection, and suppression equipment in the cable spreading rooms to determine whether any conditions or deficiencies existed which would impair the operability of that equipment.

The inspectors specifically reviewed AR 10048 for a ceiling penetration in the 'A' cable spreading room and associated ARs for other safe shutdown cable penetration electrical raceway fire barrier wrap deficiencies identified from the same licensee walkdown. The inspectors reviewed the root cause evaluation for AR 10048 to determine the significance of the barrier problem.

b. Observations and Findings

The inspectors found that AR 10048 described problems in the interfaces between floor seal penetrations and electrical raceway fire barrier wrap (gaps), for several cables required for safe shutdown of the plant. The inspectors also found that the licensee categorized AR 10048 as a significant adverse condition because it was initially determined to be reportable. The report was later retracted. The affected conduits and cables are listed in the table below. The AR documented that the fire barrier problems were identified during a walkdown to address comments made by the NRC in the Triennial Fire Protection Inspection (NRC Inspection Report No. 50-400/99-13). The inspectors found that the deficiencies were being monitored by a compensatory roving fire watch.

The inspectors found that the root cause evaluation for AR 10048 determined that other equipment would be available that was not susceptible to the postulated fire in the fire area associated with each conduit. For all but one conduit, the licensee's safe shutdown analysis (calculation E-5525, "Safe Shutdown Analysis in Case of Fire," Revision 2) had analyzed and included the other equipment as a diverse method to accomplish the same function. The other equipment credited for safe shutdown was specified in Abnormal Operating Procedure 36, "Safe Shutdown Following a Fire," Revision 9. For conduit 16159SR3 the other equipment was specified in step 14.d of Emergency Operating Procedure EPP-4, "Reactor Trip Response," Revision 7, by verifying natural circulation using the "A" train core exit thermocouples. Trending of the core exit thermocouples would provide indications of whether natural circulation existed in the reactor coolant system. Because this use of the "A" train core exit thermocouples had not been included in the safe shutdown analysis, the additional analysis was done under AR 10048. The licensee plans to incorporate this analysis of the "A" train thermocouples into calculation E-5525. The table below shows the six affected conduits and the functions affected.

AR #	Cond/Cable	Alternate Method Procedure Step	Fire Area	Function
10048	16159SR3/ 10196E	EPP-4 step 14.d	1-A-CSRB	120vac to PIC-P3 from IDP-1A-S3 circuit 3 - steam generator pressure indications PI-475, PI-485, PI-495
9922	10333A/10310A,	AOP-36, Attach. 2, step 2	1-A-EPB	FT-0122, normal charging line flow signal
	10333A/ 10333A	AOP-36, Attach 2, step 6		FT-0605A, A RHR flow indication signal and a RHR Hx bypass return flow control signal
	10333A/ 10454A	AOP-36, step 3.2.2.b		FT-940, safety injection flow indication
9922	16106D/10632D	AOP-36, Attach 1 steps 7 and 8	1-A-CSRA	120 VAC power to SSPS(A) - input IV from IDP-1B-S4 circuit 7
9922	16263P/18550Q	N/A	1-A-CSRA	spare cable - no longer required for safe shutdown
10048	16020S-SR2/ 10626A, & 10626J	AOP-36, Attach. 1, Steps 7 and 8	1-A-CSRA	120vac to SSPS A input II from IDP-1B-S2 circuit 7
		AOP-36, Attach. 1, Steps 7 and 8		RCP aux bus B UV relay & UF relay contact logic to SSPS A input II
10048	15430Q/10245A	AOP-36, Attach. 1, Step 19	1-A-BAL-B	power to MOV 2CS-L521SB-1, LCV-115E, VCT outlet isolation valve

The licensee's root cause analysis determined that the installation errors were caused by inadequate procedural guidance. The inspectors found the licensee's root cause determination to be adequate. The licensee found that the original installation procedures did not provide adequate guidance to ensure that the interface between the penetration seal and the fire barrier wrap was installed correctly. Furthermore, the licensee found that the current procedures, CMP-009, "Installation of Fire Wrap," Revision 10, and CMP-010, "Installation of Penetration Seals," Revision 12, each lacked specific guidance related to that interface and that no gap should exist.

Technical Specification (TS) 6.8.1.h states that written procedures shall be established,

implemented, and maintained covering the fire protection program implementation. Operating License Condition 2.F requires, in part, that Carolina Power and Light shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR) for the facility.

Harris FSAR section 9.5.1.2.2.1, "Safe Shutdown Capability," describes several options for ensuring that, where cable or equipment of redundant safe shutdown divisions of systems necessary to achieve and maintain cold shutdown conditions are located within the same fire area outside of primary containment, one of the redundant divisions is free of fire damage.

FSAR section 17.3 describes the Quality Assurance Program, and is implemented by NGGM-PM-007, "Quality Assurance Program Manual," Revision 4. Section 15 of the "Quality Assurance Program Manual" implements the fire protection quality assurance program. Procedures CMP-009 and CMP-010 implement the quality assurance requirements for penetration seals and fire wrap.

Contrary to the above, the licensee failed to maintain adequate procedures for the installation of electrical raceway fire barrier wrap and penetration seals in that the interface between the two would not meet the requirements of the fire protection program. The inspectors considered this failure to be a violation of Technical Specification 6.8.1.h.

Significance Determination

For each penetration where the electrical raceway fire barrier wrap and the penetration seal did not meet, thereby exposing cable for both trains in the same fire area to a fire, a diverse safe shutdown function was provided in another fire area to accomplish that same function. In addition, the use of the diverse function was called out in plant procedures. Consequently, safe shutdown could have been achieved with the available equipment.

For the six locations discussed above, the inspectors determined that the diverse safe shutdown functions met the criteria in IMC 0609, Appendix F, "Fire Protection Significance Determination Process" (SDP), for the recovery of one train. In addition, each of the fire areas affected had functional sprinkler and detection systems available. The inspectors found that each of the six locations screened out in Phase 1 of the SDP. The inspectors concluded that the risk significance of the subject violation was below the threshold for increased regulatory response, and therefore designated its risk significance as GREEN.

Enforcement

This violation is being treated as a non-cited violation, consistent with the Interim Enforcement Policy for pilot plants. This violation is in the licensee's corrective action program as Condition Report AR 10048 and is designated as NCV 50-400/00-01-01, fire protection wrap/ penetration seal interface problems.

1R09 Inservice Testing (IST)

a. Inspection Scope

For the inservice tests listed below, the inspectors reviewed the test procedures to ensure that the procedures were consistent with applicable American Society of Mechanical Engineers (ASME) code requirements. The inspectors also observed performance of the tests and/or reviewed related records, to verify that testing was being conducted in accordance with the procedures. These reviews were completed for the following IST tests:

Number	Rev.No.	Title
OST-1092	7	"1 B-SB [Residual Heat Removal] Pump Operability Quarterly Interval"
OST-1104	22	"Containment Isolation Inservice Inspection Valve Test Quarterly Interval Modes 1 - 6"
OST-1051	7	"Reactor Auxiliary Building Emergency Exhaust System Inservice Inspection Test Quarterly Interval"
OST-1190	4	"Spent Fuel Pool Cooling System Train B [Inservice Inspection] Testing Quarterly Interval Spent Fuel in the Fuel Pool(s)"

b. Observations and Findings

No findings were identified and documented through this inspection.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator performance during simulator training session EOP-SIM-17.103, "[Residual Heat Removal] Operations/Midloop Operations," Revision 0, to determine whether the operators:

- were familiar with and could successfully implement the procedures associated with placing the residual heat removal system in service, draining the reactor coolant system down to midloop level, and refilling the reactor coolant system following midloop operations;

- recognized the high-risk actions in those procedures; and

- were familiar with related industry operating experiences.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described in the Condition Reports (CRs) and Action Requests listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) with respect to the characterization of failures, the appropriateness of the associated a(1) or a(2) classification, and the appropriateness of either the associated a(2) performance criteria or the associated a(1) goals and corrective actions:

CR/AR Number	Title/Description.
CR 99-0170	E-85-B low flow trip
AR 9461	SII 7.5 KVA inverter AC input breaker trip
ARs 16223 & 16226	1MS-110 & 1MS-109 operator control tubing

b. Observations and Findings

The inspectors found that ARs 16223 and 16226 described problems with the air supply tubing for air operated condenser steam dump valves 1MS-109 and 1MS-110 identified as a result of a reactor trip on December 14, 1999. The problem was that the tubing was connected to the wrong ports which would not allow the valves to shut when the reactor coolant system average temperature decreased to 553°F and the P-12 (low-low Tave) interlock initiated a signal to shut the valves. In addition, the porting problem would have prevented main control board operation of these two valves if the P-12 interlock had been actuated. The inspectors found that the licensee had not classified this problem as a functional failure for the steam dump system.

The inspectors found that the P-12 interlock was included in Technical Specification 3.3, "Engineered Safety Features Actuation System Instrumentation," (Table 3.3-3, item 10.b) The inspectors also found that the steam dump control function was called out in step 3.d of Emergency Operating Procedure EPP-4, "Reactor Trip Response," Revision

7, to stop dumping steam (i.e. take operator action to shut the steam dumps) if average reactor coolant system temperature is below 557°F (no-load Tave) and decreasing, following a reactor trip.

The inspectors determined that the licensee's failure to identify the malfunction of the steam dump P-12 function placed the plant in a configuration in which, following a reactor trip, if the steam dumps had not shut prior to activation of P-12, the operators would not have been able to perform the EOP step of "stop dumping steam."

For the porting problem, the licensee's immediate corrective actions included correcting the tubing errors and reviewing procedure MPT-I0033, "Steam Dump Valve Testing," Revision 3, to determine why the test procedure had not identified this problem. The licensee found that the problem had not been previously identified and corrected because the P-12 function had not been tested by the procedure. The licensee subsequently revised the procedure to test that function, and issued that revision as revision 4 of MPT-I0033.

The inspectors reviewed the licensee's maintenance rule scoping document for the steam dump system, and noted that the scoping document did not explicitly include a description of the control function related to maintaining Tave at the no-load value following a reactor trip. Instead, the scoping document described the function *"remove stored energy and residual heat from primary system following reactor/turbine trip,"* The inspectors considered that this description was not clear enough to conclusively indicate that it included the control function related to maintaining Tave at the no-load value following a reactor trip. The licensee determined that this lack of clarity was a contributing reason for the failures to properly monitor the control function and to identify the functional failures. The licensee subsequently initiated AR 18158 to address not only the failure to identify the functional failures, but also the problem with the scoping document.

By reviewing ARs 16223 and 16226, the inspectors concluded that for the steam dump control function related to maintaining Tave at the no-load value following a reactor trip, the licensee either had not adequately monitored that function and/or had not adequately performed preventive maintenance as required by the maintenance rule program. The inspectors therefore considered the licensee's failure to either adequately monitor that function and/or adequately perform preventive maintenance related to that function to be a violation of 10CFR50.65 (a)(1).

Significance Determination

The inspectors considered that the failure of two steam dump valves to shut prior to 553°F would not cause an initiating event or affect the ability of a core damage mitigating system to function. In addition, a stuck open steam dump valve could be mitigated by shutting the main steam isolation valves. The inspectors concluded that the risk significance of the subject violation was well below the threshold for increased regulatory response, and have therefore designated its risk significance as GREEN.

Enforcement

Procedure ADM-NGGC-0101, "Maintenance Rule Program," Revision 11, implements 10 CFR 50.65 and provides maintenance rule implementation instructions. Section 9.2 directed personnel to obtain systems lists from the Equipment Data Base System (EDBS) and review those systems for maintenance rule applicability. For those systems identified under Section 9.2.3 and 9.2.4, each section requires that the system and function be documented in the maintenance rule database. Section 9.7, Monitoring, requires that SSC performance be monitored and evaluated against performance goals.

10 CFR 50.65(a)(1) requires, in part, that holders of an operating license shall monitor the performance or condition of structures, systems, and components (SSCs) within the scope of the monitoring program as defined in 10 CFR 50.65(b) against licensee-established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

10 CFR 50.65 (a)(2) states that monitoring as specified in (a)(1) of the rule is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function.

Contrary to 10 CFR 50.65 (a)(1) and (a)(2) , the licensee failed to set goals and monitor the performance of the steam dump system when functional failures of the low-low Tave interlock (P-12) to steam dump valves 1MS-109 and 1MS-110 occurred on December 14, 1999, which showed that the licensee had not demonstrated the performance of the system was being effectively controlled through the performance of appropriate preventive maintenance as required by 10 CFR 50.65 (a)(2).

This violation is being treated as a non-cited violation, consistent with the Interim Enforcement Policy for pilot plants. This violation is in the licensee's corrective action program as AR 18158. This violation is designated as NCV 50-400/00-01-02, failure to set goals and monitor the steam dump system under the maintenance rule.

1R13 Maintenance Work Prioritization

a. Inspection Scope

The inspectors reviewed the licensee's assessments of the risk impacts of removing from service those components associated with the emergent work items listed in Section 1R03. In addition, the inspectors reviewed AR 0016785 related to a work ticket that did not receive a risk assessment, and AR 17664 related to an active limiting condition for operation that was not identified during schedule development.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R15 Operability Evaluations

a. Inspection Scope

For the operability evaluations described in the Engineering Service Requests (ESRs) listed below, the inspectors evaluated the technical adequacy of the evaluations, to ensure that operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred:

ESR No.	Rev. No.	Title
99-00451	0	"Fire Wrap Discrepancy"
00-0012	0	"Operability During Sequencer Testing"
00-0043	0	"Main Control Room Boundary Breach"
00-00112	0	"Operability Determination for B [Emergency Service Water] Due to Leak at 1SW-139"

The inspectors' evaluation of ESR 00-00112 included also reviewing ESR 9700262, "[Emergency Service Water] Aligned to Air Compressors," Revision 0, and ESR 9400540, "[Post-Accident Sampling System] Chiller Service Water Isolation Valves," Revision 0.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R19 Post Maintenance Testing

a. Inspection Scope

For the post-maintenance tests listed below, the inspectors examined the test procedure and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately verified that the work performed was correctly completed and demonstrated that the affected equipment was functional and operable:

Test Procedure		
Number	Title	Related maintenance task
OST-1191	"Steam Generator [power operated relief valve] and Block Valve Operability Test" Rev. 5	Loop calibration of "A" steam generator power operated relief valve (PORV)

b. Observations and Findings

No findings were identified and documented through this inspection.

1R22 Surveillance Testing

.1 Surveillance Tests

a. Inspection Scope

For the surveillance tests listed below, the inspectors examined the test procedure and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately demonstrated that the affected equipment was functional and operable:

Number	Rev.No.	Title
OST-1215	22	"Emergency Service Water System Operability Train B Quarterly Interval Modes 1-2-3-4"
MST-I0175	5	"Steam Generator A Narrow Range Level Loop (L-0473) Operational Test"
OST-1049	7	"[Reactor Auxiliary Building] Emergency Exhaust System Train "B" Operability Monthly Interval Modes 1-2-3-4"

b. Observations and Findings

No findings were identified and documented through this inspection.

.2 (Closed) Licensee Event Report (LER) 50-400/1999-008-00, Control Room emergency filtration system Technical Specification noncompliance. On November 8, 1999, the licensee determined that during each of the last four refueling outages, the licensee failed to comply with Technical Specification 3.7.6, "Control Room Emergency Filtration System", in that during each outage, the licensee had blocked open the pressure boundary doors for the Control Room emergency filtration system (CREFS), to allow electrical cables to pass through the opening. Blocking the doors open had rendered both CREFS trains inoperable. With the doors blocked open the CREFS would not have been able to achieve and maintain a positive pressure in the control room, as required by Technical Specification surveillance requirement 4.7.6.d.3. During each incident, the plant was operating in mode 4.

The licensee determined that the root cause of the subject incidents was multiple failures of site personnel to recognize that blocking the control room doors open affected the operability of the CREFS. The licensee promptly addressed this cause by placing on hold the specific surveillance procedure which had been involved in the subject incidents (procedure EST-104, "Incore Thermocouple and [Resistance Temperature Detector] Cross Calibration Data Compilation," various revisions), by

providing real-time training on the subject incidents to operations and engineering personnel, and by providing site-wide training on facility changes.

The inspectors reviewed the circumstances associated with the subject incidents, and found that this LER was based on the results of the licensee's investigation of the issue as described in the Significant Adverse Condition report associated with AR 1575 (which had also been identified as CR 9803149). However, the inspectors also found that the subject of CREFS operability had been the topic of several previous corrective action program documents, as follows:

- On January 19, 1996, the licensee had initiated CR 9600140, which documented that the 10CFR50.59 safety evaluation that accompanied ESR 9500564 had been inadequate, in that although the ESR described expanding the control-room boundary beyond what was described in the licensing-basis documents, the safety evaluation had failed to recognize the change as a change to the facility as described in the licensing basis document. (The change was to include the outside vestibule door within the boundary.) The licensee's resolution of this condition included voiding the subject ESR, and initiating ESR 9600056 to replace it.

  The inspectors reviewed ESR 9600056, and noted that it effectively gave Operations personnel permission to block open control-room doors, provided that they have appropriate administrative controls to ensure that the doors are closed in the event that closing the doors became necessary. The inspectors considered that the guidance in this ESR constituted a compensatory measure to restore CREFS operability, and noted that TS 3.7.6 and the TS definition of operability do not allow operability based on compensatory measures. The inspectors therefore considered that the guidance in this ESR was not consistent with the Technical Specifications, and that the licensee's resolution of this condition was not adequate.

- On June 4, 1997, the licensee had initiated CR 9702958. This CR was initiated by a Superintendent - Shift Operations, who noted in the CR description that propping open two doors "technically makes both trains inoperable." The inspectors considered this description as an indication that a licensee employee recognized the practice as an adverse condition. However, in the corrective action program, the licensee classified this CR as an "improvement item." Subsequently, the licensee's resolution to this condition included revising procedure AP-002, "Plant Conduct of Operations," to include the inadequate guidance provided in ESR 9600056.

  The inspectors considered that the licensee's classification of this CR was inadequate, in that a condition that makes both trains of a TS-required system inoperable is clearly not an improvement item. Furthermore, because, as described above, the guidance in ESR 9600056 was not adequate, the inspectors considered that the licensee's resolution of this condition was similarly not adequate.

- On November 26, 1998, the licensee had initiated CR 9902136 (which became AR 3846). This CR was also initiated by a Superintendent - Shift Operations. It documented that two control-room doors had been blocked open for several hours, and that both trains of CREFS had been inoperable during that time. The completion notes associated with this AR stated that:

"It is reasonable to assume that as long as the door configuration is operable when the doors are shut, then shutting an open door in a reasonable amount of time would allow the [CREFS] to function in its accident capacity."

The resolution of this AR included action items for Engineering to revise EST-104, "Incore Thermocouple and RTD Cross-Calibration Data Compilation" to make it consistent with AP-002, "Plant Conduct Of Operations", for Engineering to tell Operations the allowed time for shutting any blocked-open doors, and for Operations to revise AP-002 to clarify the response expectations for the control room doors.

The inspectors noted that the licensee's resolution of this CR did not include recognizing that blocking the control-room doors open was not allowed by TS, and did not stop that practice. The residents therefore considered that the licensee failed to correct the adverse condition.

- On July 29, 1999, the licensee initiated CR 99-2136 (AR 3846), which questioned the guidance in AP-002 for CREFS operability. This CR was based partially on information contained in NRC Task Interface Agreement (TIA) 98008, "Use of Manual Compensatory Actions on Control Room Emergency Ventilation System at the McGuire Nuclear Station, Units 1 and 2," which had been issued on March 31, 1999, and of which the licensee had recently become aware. The licensee's resolution to this condition included:

  — Issuing a "night order" to Operations to obtain concurrence from the Operations Manager before breeching any emergency ventilation boundary;

  — Revising AP-002 to clarify ventilation-system boundary requirements for operability. (This action was completed in Revision 29.)

  The inspectors considered that the former item did not adequately address the issue, because the TS definition of operability does not depend on concurrence from the Operations Manager. With respect to the latter item, the inspectors reviewed AP-002, Revision 29, and noted that the licensee inserted into that procedure the statement that if the doors on both sides of a vestibule were open, both trains of ventilation would be inoperable. The inspectors also reviewed TIA 98008, and found that the conclusions expressed in that document ("It is not appropriate to take credit for manual action ... for protection of safety limits to consider equipment operable") were directly applicable to the Harris Nuclear Plant, and clearly indicated that the practice of blocking control-room doors open was not appropriate. However, the licensee's resolution of this condition did not stop the practice of blocking open the control-room doors, in that the licensee did not place procedure EST-104 on hold through the corrective actions associated with this CR.

The inspectors noted that the licensee did stop the practice of blocking open the control room doors, by placing procedure EST-104 on hold, until October 12, 1999, as a corrective action associated with AR 1575.

The inspectors considered that the licensee's practice of blocking open control-room doors to enable completion of various procedures was indeed a condition adverse to

quality. However, although the licensee had on June 4, 1997, November 26, 1998, and July 29, 1999, also identified that practice as a condition adverse to quality, the licensee had failed to initiate appropriate action to correct that condition, in that the licensee did not conclusively stop the practice until October 12, 1999. The inspectors noted that 10CFR50, Appendix B, Criterion XVI, "Corrective Action", requires, in part, that conditions adverse to quality are promptly identified and corrected. The inspectors therefore considered that the licensee's failure to promptly correct the subject adverse condition to be a violation of 10CFR50, Appendix B, Criterion XVI.

Significance Determination

The inspectors assessed the violation using NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process"(SDP), Supplement 2 and determined that the risk significance of the subject violation should be assessed using the SDP based on:

- The inspectors considered that the violation had a credible impact on safety, in that the violation allowed the licensee to periodically render inoperable both trains of a system that was specifically designed to protect the control-room staff from the radiation dose consequences of certain reactor accidents; and

- The inspectors considered that the violation involved a failure of one or more radiation barriers that could result in a significant unintended or unplanned dose, in that this violation effectively caused the failure of the CREFS, which could have resulted in a significant unintended dose to the control-room staff, if a reactor accident had occurred that involved the release of gaseous or airborne radioactive material,

  Even though the CREFS is a ventilation system, because the design function of the CREFS is to protect the control-room staff from the dose consequences of an onsite release of radioactive materials, the inspectors considered that the CREFS functioned as a barrier. Inspectable-area procedure 71111-22, "Surveillance Testing," identifies ventilation systems as barriers within the scope of that procedure. However, because IMC 0609 does not include a methodology that can be directly applied to determine the risk significance of CREFS inoperability, the inspectors considered that the violation also impacted occupational dose, in that with the CREFS inoperable, the dose to the control-room crew following a release of radioactive materials would be higher than it would be otherwise. Therefore, to assess the relative risk significance of the violation, the inspectors used the methodology described in IMC 0609, Appendix C, "Occupational Radiation Safety SDP." Using that methodology, because the violation did not involve an actual job dose > 25 person-rem, and because the violation did not involve more than 2 occurrences in an 18-month period, the inspectors determined that the violation was a GREEN finding.

Enforcement.

Because the relative significance of this violation is GREEN, we are treating this violation of 10CFR50, Appendix B, Criterion XVI, as a non-cited violation, consistent with

the Interim Enforcement Policy for pilot plants. This violation is in the licensee's corrective action program as AR18388. This violation has been designated NCV 50-400/00-01-03, inadequate corrective actions for activities rendering both trains of CREFS inoperable.

2 RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspectors reviewed the plant collective exposure history, current exposure dose trends, outage reports, "Five Year Dose Reduction Plan 1999," the year 2000 annual site dose goal, the operating chemistry plan, and historical shutdown chemistry crud bursts
and clean-up results. The inspectors also evaluated outage job evaluations and performance results, radiation work permit preparation, temporary shielding installation and removal, schedules for scaffold erection and removal, audits and self assessments, exposure tracking system, and corrective action program problem identification and resolution. The inspectors attended outage readiness planning meetings, a prejob containment entry briefing and a reactor coolant pump and pump motor change-out meeting.

b. Observations and Findings

No findings were identified and documented through this inspection.

4 OTHER ACTIVITIES

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on April 6, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

NRC

B. Bonser, Chief, Reactor Projects Branch 4
R. Laufer, Harris Project Manager, NRR

Licensee

D. Alexander, Regulatory Affairs Manager
K. Altman, Major Projects Manager
C. Burton, Site Operations Director
R. Duncan, Harris Plant General Manager
R. Field, Nuclear Assessment Manager
T. Hobbs, Operations Manager
J. Holt, Outage and Scheduling Manager
G. Kline, Harris Engineering Support Services Manager
T. Natale, Training Manager
K. Neushaeffer, Plant Support Services Manager
J. Scarola, Harris Plant Vice President
B. Waldrep, Maintenance Manager
E. Wills, Environmental & Radiation Control Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-400/00-01-01	NCV	Fire protection wrap/penetration seal interface problems (Section 1R05)
50-400/00-01-02	NCV	Failure to set goals and monitor the steam dump system under the maintenance rule (Section 1R12)
50-400/00-01-03	NCV	Inadequate corrective actions for activities rendering both trains of CREFS inoperable (Section 1R22.2)

Previous Items Closed

50-400/1999-008-00	LER	Control room emergency filtration system Technical Specification noncompliance (Section 1R22.2)

Previous Items Discussed

None

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
● Initiating Events	● Occupational	● Physical Protection
● Mitigating Systems	● Public	
● Barrier Integrity		
● Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent little effect on safety. WHITE findings indicate issues with some increased importance to safety, which may require additional NRC inspections. YELLOW findings are more serious issues with an even higher potential to effect safety and would require the NRC to take additional actions. RED findings represent an unacceptable loss of safety margin and would result in the NRC taking significant actions that could include ordering the plant shut down.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. The color for an indicator corresponds to levels of performance that may result in increased NRC oversight (WHITE), performance that results in definitive, required action by the NRC (YELLOW), and performance that is unacceptable but still provides adequate protection to public health and safety (RED). GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, as described in the matrix. The NRC's

actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.