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

January 28, 2002

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, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT NO. 50-400/01-05

Dear Mr. Scarola:

On December 29, 2001, the Nuclear Regulatory Commission (NRC) completed an inspection at your Shearon Harris reactor facility. The enclosed report presents the results of that inspection which were discussed on January 7, 2002, with you and other members of your staff.

The inspection examined 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). One of these issues was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Shearon Harris facility.

Immediately following the terrorist attacks on the World trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of Carolina Power and Light's response to these advisories and Harris' ability to respond to terrorist attacks with the capabilities of the current design basis threat. From these audits, the NRC has concluded that the Harris security program is adequate at this time.

In accordance with 10CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report

cc w\encl: (See page 3)

cc w/encl:
Terry C. Morton, Manager
Performance Evaluation and
Regulatory Affairs CPB 9
Carolina Power & Light Company
Electronic Mail Distribution

Chris L. Burton
Director of Site Operations
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

Robert J. Duncan II
Plant General Manager--Harris Plant
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

Richard J. Field, Manager
Regulatory Affairs
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

John R. Caves, Supervisor
Licensing/Regulatory Programs
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

William D. Johnson
Vice President & Corporate Secretary
Carolina Power & Light Company
Electronic Mail Distribution

John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, NW
Washington, DC 20037-1128

Mel Fry, Director
Division of Radiation Protection
N. C. Department of Environmental
Commerce & Natural Resources
Electronic Mail Distribution

Peggy Force
Assistant Attorney General
State of North Carolina
Electronic Mail Distribution

Public Service Commission
State of South Carolina
P. O. Box 11649
Columbia, SC 29211

Chairman of the North Carolina
Utilities Commission
P. O. Box 29510
Raleigh, NC 27626-0510

Robert P. Gruber
Executive Director
Public Staff NCUC
P. O. Box 29520
Raleigh, NC 27626

Vernon Malone, Chairman
Board of County Commissioners
of Wake County
P. O. Box 550
Raleigh, NC 27602

Richard H. Givens, Chairman
Board of County Commissioners
of Chatham County
Electronic Mail Distribution

Distribution w/encl: (See page 4)

Distribution w/encl:
J. Goshen, NRR
C. Evans (Part 72 Only)
RIDSNRRDIPMLIPB
PUBLIC

OFFICE	DRP/RII	DRP/RII	DRP/RII	DRS/RII	DRS/RII	DRS/RII	
SIGNATURE	GMacDonald:vg	JBrady	BHagar	SVias	WRogers	ETesta	
NAME	gm	jb	bh	sv	wr	et	
DATE	1/28/2002	1/28/2002	1/28/2002	1/28/2002	1/28/2002	1/28/2002	
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	

OFFICIAL RECORD COPY DOCUMENT NAME: C:\Program Files\Adobe\Acrobat 4.0\PDF Output\0105 drp.wpd

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	50-400/01-05
Licensee:	Carolina Power & Light (CP&L)
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	September 30 - December 29, 2001
Inspectors:	J. Brady, Senior Resident Inspector R. Hagar, Resident Inspector S. Vias, Senior Reactor Inspector (Sections 4OA5.6 through 4OA5.16) E. Testa, Senior Health Physicist (Sections 2OS1, 2OS2 & 4OA1) G. MacDonald, Senior Project Engineer (Sections 1R13.2, 1R17, & 1R23) W. Rogers, Senior Reactor Analyst (Section 1R13.4)

Approved by: B. Bonser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000400-01-05, on 9/30 - 12/29/2001, Carolina Power & Light, Shearon Harris Nuclear Power Plant, Unit 1. Fire Protection, Maintenance Risk Assessments and Emergent Work Evaluation, and Identification and Resolution of Problems.

The inspection was conducted by resident inspectors, a region based Senior Project Engineer, a Senior Reactor Engineer, a Senior Health Physicist, and a Senior Reactor Analyst. The inspection identified two Green findings, one of which was a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. Two examples of failing to implement the fire protection program were identified in the B cable spreading room (CSRB). The first example involved the failure to have automatic sprinklers in the CSRB tunnel area where multiple safety-related cable trays contain safe shutdown cables. The second example involved a failure to follow the design control program for resolution of unqualified thermolag fire barriers.

  The safety significance was determined to be very low because of the very low probability of a fire in this area, and because of proceduralized operator recovery actions that would restore off-site power to the one safety bus that potentially would be lost. (Section 1R05)

- Green. A finding was identified in Inspection Report 50-400/00-04 related to an inaccurate risk assessment for the B Startup Transformer outage that occurred in July 2000. The inadequate risk assessment was due to an error in the risk assessment model.

  The safety significance has been determined to be very low because compensatory actions were put in place to be able to return the transformer to service in two hours, and because the risk reduction associated with those compensatory actions offset the risk increase caused by the inaccurate risk assessment. (Section 1R13.4)

Identification and Resolution of Problems

- The inspectors concluded that collectively the inspection findings indicated that the corrective action program was having a positive impact on risk reduction, but some issues were still being identified due to events and NRC inspections. (Section 4OA2)

B. Licensee Identified Violations

None.

Report Details

Prior to the beginning of this inspection period, the unit had been shutdown and the reactor fuel had been offloaded as part of a refueling outage. Fuel was reloaded on December 3. The unit remained shutdown through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's preparations for cold weather as described in procedure AP-301, "Adverse Weather," Revision 30, to verify that those preparations adequately protected accident mitigation systems from adverse weather effects. The inspectors selected for inspection the following two systems that are required to be protected from adverse weather:

- Emergency Service Water/Screen Wash Functions
- Refueling Water Storage Tank

For each of these systems, the inspectors reviewed the preparations to verify that the protection features were monitored; that operator actions defined in procedure AP-301 maintain system readiness; and that the system could perform its safe shutdown function during the anticipated weather conditions. This included review of performance of the following procedures:

- ORT-1415, "Electric Unit Heater Check Monthly Interval," Revision 5
- OP-161.01, "Operations Freeze Protection and Temperature Maintenance Systems," Revision 13
- Work Order 88809-01 which implemented PIC-E048, " Heat Tracing Control Temperature and Readout Unit Calibration," Revision 17
- Work Order 193837-02 which implemented maintenance checklist CL-E0010, "Heat Trace Panel Current Check and Relay CSR-4A Calibration"
- Work Order AKLE 003 (AMMS) which implemented maintenance checklist CL-I0008, "Temperature Switch"

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial Walkdown

For the system identified below, the inspectors reviewed the identified plant documents to determine correct system lineup, and observed equipment to verify that the system was correctly aligned:

B train spent fuel pool cooling with the A train spent fuel pool cooling emergency power source (A emergency diesel generator) out of service and the reactor core off-loaded to the A spent fuel pool on October 19, 2001

- Procedure OP-116, “Spent Fuel Pool System,” Revision 22
- Drawing 2165-S-0805, “Simplified Flow Diagram Fuel Pool Cooling System,” Revision 8

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Within the areas identified below, the inspectors observed the following to determine whether any conditions adversely affected fire protection defense-in-depth features:

- transient combustible materials;
- any welding or cutting being performed in the area;
- the physical condition of the fire detection devices;
- the physical condition of the automatic suppression system (where used);
- the availability and general condition of portable fire extinguishers;
- the physical condition of manual suppression systems, including fire hoses and hose stations;
- the material condition of electrical raceway fire barrier systems;
- the material condition of the fire door(s);
- the condition of ventilation fire dampers;

- the material condition of the structural steel fire-proofing (where used);
- the physical condition of seals in accessible electrical and piping penetrations; and
- the adequacy of compensatory measures, where degraded features were identified.

The inspected areas include the following:

- “A” switchgear room
- “B” switchgear room
- cable spreading room
- “A” chiller area
- “B” chiller area
- control room
- fuel handling building 236' elevation

The following procedures were reviewed:

- AOP-25, “Loss Of One Emergency AC Bus (6.9kV) Or One Emergency DC Bus (125 VDC),” Revision 20
- AOP-36, “Safe Shutdown Following a Fire,” Revision 18

b. Findings

Two examples of a violation were identified for failing to implement the fire protection program in the B cable spreading room (CSRB) (Green). The first example involved the failure to have automatic sprinklers in the CSRB tunnel area where multiple safety-related cable trays exist that contain safe shutdown cables. The second example involved a failure to follow the design control program for resolution of unqualified thermolag walls.

The inspectors found that the cable spreading room B cable tunnel, located in cable spreading room A, did not have automatic fire suppression (sprinklers), although Final Safety Analysis Report (FSAR) section 9.5.1 indicates that all of the cable spreading rooms have automatic suppression. The inspectors found that Engineering Service Request (ESR) 95-00620, “Thermolag Fire Protection Issues Resolution,” Revision 1, acknowledged the lack of sprinklers in this area. The cable spreading rooms have had a compensatory fire watch since thermolag problems were identified in November 1999 during the NRC Fire Protection Inspection (Inspection Report 50-400/99-13).

10 CFR 50.48 requires that all operating nuclear power plants have a fire protection program that satisfies Criterion 3 of Appendix A to 10 CFR 50. Harris Operating License NPF-63, Condition 2.F, “Fire Protection Program,” requires that the fire protection program described in the FSAR be implemented and maintained in effect as approved

in the Safety Evaluation Reports (SERs). FSAR section 9.5A, Fire Protection Hazards Analysis, describes the CSRB as having automatic suppression. Section 9.5.1 of NUREG 1038 (SER), described that the basis for NRC acceptance of the fire protection plan in the cable spreading rooms was, in part, that the primary fire suppression system was an automatic pre-action sprinkler system with fusible-type sprinkler heads. Contrary to the above the CSRB cable tunnel has not had automatic suppression since the time of plant licensing. The fire hazards analysis in FSAR section 9.5A does not reflect the lack of automatic suppression, nor has the NRC accepted a deviation from Branch Technical Position (BTP) CMEB 9.5-1 in an SER. The condition represents the licensee's failure to follow their Fire Protection Program as required by the operating license.

While assessing the risk significance of not having sprinklers, the inspectors found another problem. The inspectors observed that the CSRB tunnel is enclosed on either one or two sides, depending on the location, by thermolag fire barrier material. The other sides are concrete three hour fire walls. The FSAR indicates that the thermolag was rated as adequate for the hazard, instead of as a 3 hour barrier, due to changes initiated by ESR 95-00620, "Thermolag Fire Protection Issues Resolution," Revision 1, to resolve thermolag barrier issues. The ESR and associated calculation FP-0110, "Evaluation of Thermo-lag Fire Barrier Enclosures Within the Cable Spreading and ACP Rooms," Revision 0, relied on an assumed 1" air gap on one side of the vertical thermolag wall to mitigate the consequences of the thermolag fire barrier having less than a 3 hour fire rating.

The inspectors found that the 1" air gap was identified as an assumption in the calculation but not identified as an assumption in the ESR, and consequently, was not validated as appropriate and correct as required by Procedure EGR-NGGC-0005, "Engineering Service Requests," Revision 5. Therefore, the assumption was an unverified assumption. Although the ESR indicated that the air gap was verified to exist, it acknowledged that a direct conducting path between the thermolag wall and the cable trays would invalidate the function of the 1" air gap. The inspectors looked for either physical or administrative protection of the air gap such that direct conducting paths could be prevented, but none were found. The inspectors noted that recent modification activities which installed additional cable (for the additional spent fuel pools) in the CSRB tunnel cable trays had temporarily hung rolls of cable off of the trays in close proximity to the CSRB thermolag wall during the installation process. The inspectors did not find the 1" air gap mentioned in any FSAR descriptions of the barrier for the cable spreading room fire areas. Also, there were no design drawings that showed the inclusion of the 1" air gap as part of the fire barrier because the ESR did not include changing the drawings. The inspectors observed that although the licensee assessed the affect that an air gap would have in reducing the damage from a fire that impacted the thermolag fire wall, the air gap was never included in the plant design, or controlled, as a necessary design feature. The inspectors concluded that the unverified assumption had not been properly validated as required by the procedure.

FSAR section 17.3 indicated that the QA Program applies to fire protection and is implemented through the Quality Assurance Program Manual, and implementing procedures. The Corporate Quality Assurance Program Manual, Sections 15 and 3 required that design changes for fire protection related items be prepared in accordance with the design control program. Procedure EGR-NGGC-0005, Section 9.3 requires that assumptions are to be clearly identified and documented, and that assumptions are to be validated to confirm the proposed change and/or conceptual solution are appropriate and correct. Unverified assumptions shall be validated via inspection, demonstration, and/or analysis to verify that the modified system/component functions/performs as intended; the design change has been correctly implemented; and, the revised design is correct. Contrary to the above, the licensee did not validate that the use of the air gap was appropriate and correct. The condition represents a second example of the licensee’s failure to follow their Fire Protection Program as required by the operating license.

Significance

The cable trays in CSRB contain safety-related cables that perform various safe shutdown control functions associated with mitigating system equipment. Some of the cables that provide the same redundant safe shutdown train functions are located on the other side of the thermolag wall in CSRA. Consequently, a fire in the CSRB tunnel that would impact the thermolag wall and exist for greater than 1.8 hours would impact the cables in CSRB and could impact the redundant function cables in CSRA. The postulated fire scenario in the CSRB tunnel could lead to the following initiating events:

- a reactor trip with loss of the power conversion system,
- a stuck open power operated relief valve (PORV),
- loss of off-site power with loss of one division of alternating current (AC), and
- a reactor coolant pump seal failure /small break loss of coolant accident.

The failure to have sprinklers in the CSRB tunnel and the failure to follow the design control program for the thermolag walls was collectively determined to have very low safety significance (Green) because of the very low probability of a fire in this area, and because of proceduralized operator recovery actions that would restore off-site power to the one safety bus that would be potentially lost.

The failure to have automatic fire suppression (sprinklers) in the CSRB tunnel is in the licensee’s corrective action program as Action Request (AR) 51820. The failure to follow the design control program for a modification to fire protection features as required by the FSAR and their Corporate Quality Assurance Manual is in the licensee’s corrective action program as AR 46469. These two examples of failing to implement the fire protection program as required by license condition 2H have been designated as a Non-cited Violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. It is identified as NCV 50-400/01-05-01, Two examples of Failure To Implement The Fire Protection Program In B Cable Spreading Room Tunnel.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed the inspection of the 1A component cooling water heat exchanger. This was an inspection that was not previously planned for the outage, but was added due to a differential pressure alarm received immediately prior to the outage. The inspectors reviewed licensee records to verify that acceptance criteria and results appropriately considered differences between testing conditions and design conditions, that inspection results were appropriately categorized against pre-established acceptance criteria, that frequency of testing or inspection was sufficient to detect degradation prior to loss of heat removal capability below design basis values, and that test results considered test instrument inaccuracies and differences.

Due to fouling found during the inspection the licensee initiated additional inspections. The inspectors reviewed the following ARs associated with the additional inspections to determine whether the licensee identified appropriate corrective actions:

AR Number	Title/Description
47942	Service water side fouling of CCW heat exchanger and A EDG jacket water heat exchanger fouling
47584	Service water side fouling of ESCW heat exchanger
50393	Inadequate service water flow to the A CSIP oil coolers
50611	B EDG jacket water heat exchanger fouling
50768	Service water system performance

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors reviewed licensed operator requalification simulator training on October 9. This observation included emergency operating procedure (EOP) and abnormal operating procedure (AOP) scenarios. The training was completed in accordance with the following Exercise Guides:

Guide	Title	Revision
EOP-SIM-18.02	“Loss of [Heater Drain Pumps] / Loss of [Main Feed Water] Pump / [procedure] EPP-004 / [procedure] EDP-005"	0
EOP-SIM-18.03	“Loss of Heat Sink”	0

The scenarios tested the operators’ ability to perform appropriate actions for a loss of both heater drain pumps, a loss of main feed water pump, a loss of a feedwater train, and a loss of all feedwater to one steam generator. In addition, the scenarios tested the operators’ ability to stabilize the plant following a reactor trip, and to cooldown the plant using natural circulation. The inspectors focused on clarity and formality of communication, use of procedures, alarm response, control board manipulations, group dynamics and supervisory oversight.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described in the ARs 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:

AR Number	Subject/Description.
45543	Failure of inlet damper to E-13 fuel handling building emergency exhaust
46186	Breaker for air handler-4B tripped when started in high speed
44662	B primary makeup water pump failure to start
48376	1RH-39 failure to operate for shutdown cooling
45763	A Hydrogen analyzer problems
50736	Reactor coolant system flow bistable FB-434A failure to function properly

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

.1 Outage Risk Assessment

a. Inspection Scope

Periodically during the outage, the inspectors reviewed the licensee's risk assessments for the schedule changes made during the outage, to verify that those assessments were being completed in accordance with procedure OMP-003, "Outage Shutdown Risk Management," Revision 14. The inspectors reviewed plant configurations to verify that those configurations were consistent with the risk assessments and to verify that key safety functions were being preserved.

b. Findings

No findings of significance were identified.

.2 Foreign Material Found in Containment Sump Suction Line to Residual Heat Removal Pump A

a. Inspection Scope

The inspectors reviewed the licensee's risk assessment and the emergent work evaluation associated with the discovery of foreign material in the Emergency Core Cooling System (ECCS) suction piping. AR 49404 was reviewed to determine whether the licensee identified and implemented appropriate corrective actions. The following documents were reviewed:

- AR 49404, Foreign Material Found In Line Downstream of Valve 1SI-310, Revision (Rev.) 0
- Licensee Event Report (LER) 50-400/2001-003-00, 1A-SA Residual Heat Removal (RHR) Suction Line Debris - Nonconforming Condition, Rev. 0
- ESR 01-00201, HNP Risk Estimation for Debris Near 1SI-310, Rev. 0
- ESR 01-00207, Past Operability Evaluation of Debris Found in A CT Suction (containment spray)
- Procedure Maintenance Management Manual (MMM)-011, Cleanliness and Housekeeping, Rev. 4
- Work Order 89715-01

b. Findings

A failure of the licensee’s foreign material exclusion (FME) controls was identified when several pieces of foreign material were discovered in the containment sump suction piping to the A RHR pump. This item was unresolved pending completion of the significance determination.

On October 8, 2001, while performing maintenance to repair a body-to-bonnet leak on 1SI-310, containment vessel sump to RHR pump1A-SA downstream isolation valve, licensee mechanics observed foreign material in the RHR system piping consisting of a plastic cable tie and several pieces of rubber material. Further inspection found additional debris. The largest was a piece of rubber material approximately 5"X20"X3/16". The material was removed and the licensee initiated AR 49404 and a significant adverse condition investigation to determine the root cause and to implement corrective action. Further inspection found debris in containment spray sump suction piping as well. Subsequent inspections by the licensee included RHR and containment spray system suction piping, the containment recirculation sump piping, the refueling water storage tank and containment spray discharge lines near containment isolation valves, and the A RHR pump impeller eye. Further discussion of the extent of condition evaluation is included in sections 4OA2 and 4OA3 of this report. The licensee determined that the root cause of the debris in the containment sump suction piping was historical poor work practices with respect to FME control. The investigation team could not determine conclusively when the foreign material was introduced, however the most probable time was during work on the A RHR pump in 1991.

An engineering analysis concluded that the pump flowrate would be degraded due to partial blockage of the pump suction. The largest piece of material could block approximately 60 percent of the flow area. The inspectors verified that at 60 percent flow blockage the RHR pump would have adequate NPSH for proper operation. Additionally, the inspectors verified that the 60 percent degraded flow was adequate for the reduced flow requirements necessary for the high pressure recirculation function. An operability determination for past performance concluded that the A RHR pump would have been inoperable in the low pressure recirculation mode. The debris would not have affected the pump in the normal RHR shutdown cooling or accident injection modes due to the different flowpaths. The material in the containment spray system did not affect system operability.

Technical Specification (TS) 6.8, Procedures and Programs, section 6.8.1 requires that written procedures be established implemented and maintained covering the activities recommended in Appendix A of Regulatory Guide (RG) 1.33, Revision 2, February 1978 which includes section 9.0, Procedures for Performing Maintenance. Licensee Maintenance Management Manual (MMM) Procedure MMM-011, “Cleanliness and Housekeeping,” Revision 4, section 5.3,” Preventing Contamination During Maintenance,” contains the requirements to prevent foreign object entry into plant systems and components.

Contrary to the above, adequate foreign material exclusion controls were not implemented for the RHR System when on October 8, 2001, foreign material of a size to affect pump performance (greater than the containment sump screen openings) was identified in the containment sump suction piping to the A RHR pump. The exact time of entry of this foreign material could not be established. The most probable time was during work on the A RHR pump during maintenance in 1991. This item has been identified as an Unresolved Item (URI) 50-400/01-05-02, Foreign Material in A RHR Containment Sump Suction Piping, pending completion of the significance determination.

.3 Failure of 1RH-39 to Open

a. Inspection Scope

The inspectors examined the circumstances associated with the failure of 1RH-39 to open on September 22 to determine if the plant had been operating with a degraded condition and, if so, for how long. (This valve is one of two isolation valves between Reactor Coolant System loop C and the RHR pump suction header. This failure is mentioned in section 1R20 of this report.) The inspectors also reviewed both the AR initiated by the licensee to document this failure (AR 48376, “Valve 1RH-39 Failed to Stroke”) and the corresponding Significant Adverse Condition investigation report.

b. Findings

The inspectors identified a failure to follow work instructions in terminating a control circuit cable lug to a post in valve 1RH-39 breaker cabinet. The valve failed to open on September 22, when the licensee attempted to initiate shutdown cooling. This item was unresolved pending completion of the significance determination.

The licensee’s diagnosis of the failure revealed that 1RH-39 had failed to open because one of the lugs that connected a cable in its control circuit to a post in its breaker cabinet had not been properly landed, and had worked loose enough to disable the control circuitry. The valve was restored to operation after the licensee properly landed the subject lug, approximately four hours after the failure occurred.

The licensee’s investigation revealed, in part, that the subject lug had last been the object of work activities completed in September 1995, when work request/job order (WR/JO) 94-AJLP1 had been initiated and completed to replace the terminal blocks in the breaker cabinet, for Plant Change Request (PCR) 7167. The licensee also determined that work performed under the authority of that work request had not properly landed the subject lug. The inspectors’ review of the subject WR/JO revealed that it included instructions to not only replace the subject terminal blocks, but also to independently verify that all verifiable components, including wires and fuses, were properly aligned. The inspectors noted that the WR/JO did not explicitly require personnel to properly land lugs, but considered that properly landing lugs was within the skill of the mechanics’ craft. The inspectors thus considered that the instruction to replace the subject terminal blocks included the implied instruction to properly land the associated lugs. Therefore, the inspectors considered that the failure to properly land the subject lug and to independently verify proper alignment of all verifiable components was a failure to follow the instructions in the subject WR/JO.

The licensee's failure to follow the subject work instructions was not a minor violation because it had an actual impact on safety, in that it resulted in the loss of one train of shutdown cooling. This degradation of one train of long-term heat removal affected the Mitigation Systems cornerstone, and it represented an actual loss of one train of a safety function of equipment designated as risk-significant in accordance with the Maintenance Rule, for longer than 24 hours.

10 CFR 50, Appendix B, Criterion V ("Instructions, Procedures, and Drawings") requires in part that activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, of a type appropriate to the circumstances. To implement the change described in PCR 7167, that replaced the terminal blocks in the breaker cabinet for 1RH-39, documented work instructions were included in work request WR/JO 94-AJLP1. The licensee failed to comply with the work instructions in WR/JO 94-AJLP1 in that the licensee failed to properly replace the subject terminal blocks, and failed to independently verify that all verifiable components, including wires and fuses, were properly aligned. This item is designated as URI 50-400/01-05-03, Failure to properly terminate a lug in the control circuit of motor-operated valve 1RH-39, pending completion of the significance determination. This issue is documented In the licensee's corrective action program as AR 48376.

.4 (Closed) Finding (FIN) 50-400/00-04-01; Inaccurate Risk Assessment of Startup Transformer. Section 1R13 of Inspection Report 50-400/00-04 identified that an inaccurate risk assessment had been performed for the B Startup Transformer outage that occurred in July 2000. The inadequate risk assessment was due to an error in the risk assessment model. The NRC completed a Phase III risk evaluation of the performance deficiency, and concluded that the deficiency was of very low safety significance (Green). The licensee's invalid risk assessment for taking the startup transformer out of service indicated that the risk increase was so small that compensatory measures were not needed. However, the licensee did take compensatory measures to be able to credibly return the transformer to service within two hours throughout the maintenance evolution. The compensatory actions significantly reduced the risk associated with performing the maintenance, and effectively offset the risk increase caused by the inaccurate risk assessment. Because this occurred before 10 CFR 50.65(a) (4) was in effect, this item was not a violation, but was identified as a finding.

1R15 Operability Evaluations

a. Inspection Scope

For the operability evaluations described in the 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/AR No.	Rev. No.	Title/Description
ESR 01-00160	0	[Emergency Diesel Generator] 1A-SA Fuel Pump Base to Block Mounting Cap Screw Failure
ESR 01-00177	0	Operability assessment for reactor vessel calculated sub-region volume discrepancies
ESR 01-00207	0	Past Operability Evaluation of Debris Found in A Containment Spray Suction
AR 50708	0	Diaphragm Leaks on All 3 Pressurizer Power-Operated Relief Valves

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed selected portions of the following modifications to ensure that the planned testing would verify that selected safety functions which had been affected by the modification would be demonstrated to be operable.

- ESR 97-00807, Revision (Rev.) 2, Large Bore Piping, Main Feedwater, Main Steam, Auxiliary Feedwater.
- ESR 97-00537, Rev. 0, Main Feedwater Tempering Valve Removal
- ESR 00-00262, Rev. 2, Steam Generator Replacement/Power Uprate Instrumentation Changes

For ESR 97-00807 and 97-00537 the inspectors reviewed the following documentation to ensure that the proposed testing would adequately verify that the feedwater isolation safety function which had been affected by the steam generator replacement project piping removal was demonstrated to be operable.

- Drawing CPL-2165 S-0544, Rev. 33
- Drawing 1364-47239, Rev. 4
- Drawing 2166 B401 sheet 1835, Rev. 7
- Drawing 2166 B401 sheet 1917, Rev. 9
- Drawing 1364 2776 S30 Sheet 1, Rev. 12
- Operations Surveillance Test (OST)-1844, “Slave Relay Component Operability Verification Refueling interval,” Rev. 7
- OST-1853, “Feedwater Isolation ESF Response Time Trains A and B 18 Month Interval,” Rev. 9
  For ESR 00-00262 the inspectors reviewed the following documents to ensure that the proposed testing would adequately verify the main steam flow loop

instrumentation safety functions:

- Process Instrumentation Control Scaling Calculation, SC-N-112, Rev. 6
- Maintenance Surveillance Test (MST)-I0017, Main Steam / Feedwater Flow Loop 1 (F-0475/F-0476) Channel Calibration, Rev. 8
- MST-I0135, Main Steam Feedwater Flow Loop 1 (F-0475/F-0476) Operational Test, Rev. 6

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

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

Test Procedure		
Number	Title	Related maintenance task
OST-1085	"1A-SA Diesel Generator Operability Test Semiannual," Revision 16	Disassembly of 1A-SA emergency diesel generator and replacement of engine cam shafts.
OST-1013	"1A-SA Emergency Diesel Generator Operability Test Monthly Interval Modes 1-2-3-4-5-6," Revision 16	Major maintenance on the 1A-SA emergency diesel generator (during the outage)
OST- 1103	"Component Cooling Water ISI Valve Test Refueling Interval," Revision 11	Motor operated valve maintenance
OST-1214	"Emergency Service Water System Operability Train A Quarterly," Revision 24	Motor operated valve maintenance
OST-1805	"Pressurizer [Power-Operated Relief Valve] Operability 18 Month Interval," Revision 10	Maintenance on Pressurizer Power-Operated Relief Valve actuators
OST-1216	"Component Cooling Water System Operability (A-SA and B-SB Pumps In Service)Quarterly," Revision 15	Valve maintenance

OST-1842	"Turbine Trip: ESF Response Time Train A or B 18 Month Interval," Revision 10	Main Turbine Disassembly and Outage Maintenance
OST-1043	"Reactor Coolant System Vent Path Operability Quarterly Interval," Revision 11	Maintenance on vent valves
EST-212	"Type C Local Leak Rate Tests," Revision 31	Maintenance on isolation valves for penetrations M-9, M-10, and M-11

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

Refueling Outage 10 (RFO-10) started on September 22. The following is a description of the scope of inspections performed for refueling and outage-related activities:

- At various times during the outage, the inspectors examined the site to verify that the configuration-specific Key Safety Function Availability Checklist was posted at conspicuous spots throughout the plant, including the main control room. The inspectors routinely reviewed licensee activities to confirm that the licensee followed the outage risk control plan and maintained operable the systems that provided the key safety functions.

- On a random-sampling basis, the inspectors reviewed current clearance tags to verify that the tags were properly hung and that associated equipment was appropriately configured to support clearance functions.

- The inspectors reviewed fuel handling operations to verify that those operations and related activities were being performed in accordance with TS and the following procedures:

Number	Title	Revision
FMP-106	"New Fuel Receipt inspection and Storage Location Verification"	13
FHP-020	"Refueling Operations"	20
FHP-014	"Fuel and Insert Shuffle Sequence"	24

The inspectors reviewed licensee activities to verify that the licensee was tracking movement of fuel assemblies (including new fuel assemblies), from core offload through core reload. The inspectors specifically reviewed licensee activities related to new fuel assemblies numbered HN33 and HN54, and offloaded fuel assemblies HM01, HM23, & HM58.

After the core was reloaded, the inspectors reviewed the licensee's use of a remote-controlled camera to verify proper fuel-assembly positioning, in accordance with procedure FHP-010, "Core Mapping Following Fuel Loading," Revision 10.

- For changes in the unit's operational mode, the inspectors reviewed licensee activities on a sampling basis to verify that TS requirements and prerequisites from procedures GP-002, "Normal Plant Heatup from Cold Solid to Hot Subcritical, Mode 5 to Mode 3," Revision 21; were met prior to the mode changes. Prior to reactor startup, the inspectors examined areas inside the containment building to verify that debris had not been left which could affect performance of the containment sumps, and that the licensee's performance of procedure OST-1081, "Containment Visual Inspection When Containment Integrity Is Required," Revision 8, was adequate.

b. Findings

While observing portions of the plant cool-down to Mode 5, the inspectors observed that when the operators attempted to place the RHR system into service, motor-operated valve 1RH-39 failed to open. That failure is described in section 1R13.3.

1R22 Surveillance Testing

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.	Title
OST-1809*	11	"Switchover to Recirculation Sumps, ESF Response Time 18 Month Interval"
OST-1824	21	"1B-SB Emergency Diesel Generator Operability Test 18 Month Interval"
OST-1812	14	"Auxiliary Feedwater Isolation ESF Response Time 18 Month Interval"
OST-1808	9	"Main Steam Isolation: ESF Response Time 18 Month Interval"
OST-1830	9	"Turbine Driven Auxiliary Feedwater Pump Auto Start: ESR Response Time Train A 18 Month Interval Mode 3-4"

*This procedure included inservice testing requirements.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modification ESR 01-00055, Temporarily Modify the A Reactor Coolant Pump (RCP) Circuits to Support Steam Generator (S/G) Replacement, Revisions 0 through 4, to ensure that the planned post modification testing would verify that the RCP circuit safety functions would be demonstrated to be operable.

The following documents were reviewed:

- Drawing CAR 2166 B-401 sheet 110, Rev. 9
- Drawing CAR 2166 B-401 sheet 126, Rev.1
- Drawing CAR 2166 B-401 sheet 102, Rev. 10
- Procedure MST-E0074, RCP Undervoltage Relay Calibration, Rev. 2
- Procedure OST-1125, RCP A Undervoltage and Underfrequency TADOT Quarterly Modes 1-5, Rev. 8
- Procedure OST-1044, Engineered Safety Features Actuation System Train A Slave Relay Testing Quarterly Modes 1-4, Rev. 19
- Action Request 00050540, ESR 01-000540 Inadequate Restoration Testing
- Procedure EGR-NGGC-0155, "Specifying Electrical/I&C Modification Related Tests," Rev. 1

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspectors evaluated radiological surveys and access controls to verify their implementation for Refueling Outage 10 (RFO 10), and to verify that work was conducted in accordance with Radiation Work Permits (RWP). ARs 46144, 47476, 47760, and 48775 were evaluated for assignment of responsibility, resolution and timely closure.

Pre-job briefings, work-in-progress, and health physics (HP) technician job coverage were observed for RWP 0140, Installation of Leak Control Flange on Transfer Tube. The inspectors observed steam generator replacement work in progress on the closed circuit television (CCT) monitors. The inspectors toured containment and observed access controls for Very High Radiation Areas (VHRAs). Personnel dosimetry results and exposure investigation reports were independently evaluated and discussed for Uptakes and Personnel Contamination Events. The inspectors observed access control measures for “rolling radiation areas” created by the movement of the old steam generators from containment to the entombment building. Licensee radiation protection and access control activities were evaluated against FSAR, TS, and 10 CFR Part 20 requirements.

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

a. Inspection Scope

The inspectors evaluated the plant collective exposure history, current exposure dose trends, and the year 2001 annual site dose goal to determine if the licensee was implementing ALARA practices as required by 10 CFR 20.1101(b). The inspectors evaluated procedures, Radiation Work Permits for the outage and ALARA Work Plan Dose Estimates. The inspectors attended several ALARA Committee meetings that reviewed, discussed and approved dose estimate changes for work activities.

The inspectors evaluated the following ALARA Work Plans for RFO 10 for lessons learned, dose goal planning and job dose tracking:

Radiation Work Permit Number(s)	Title
01-001	RFO-10 Refueling Activities
01-002	Seal Table Maintenance Activities
01-003	Reactor Coolant Maintenance
01-012 & 01-205	Install/Remove Insulation
01-013 & 01-203	Shielding Installation and Removal
01-011 & 01-206	Installation and Removal of Scaffold
01-207	Steam Generator Replacement Reactor Coolant System Severance/Machine/Weld/Primary Foreign Object Search and Retrieval
01-209	Rig Out Old Steam Generator/Transport/Rig in New Steam Generator

Procedures reviewed included the following:

- HPS-NGGC-0003, "Radiological Posting, Labeling and Surveys," Revision 5
- HPP-625, "Performance of Radiological Surveys," Revision 15

During the containment tour, scaffolding installation activities were observed and ALARA practices evaluated. Independent surveys were performed of the "rolling radiation area" during steam generator movement. The inspectors evaluated the route used for steam generator transportation and evaluated the contingency plans for a dropped component. Actual job dose was compared to estimated job dose for the steam generator movement and for the spent fuel cask basket transfer. Pre-job ALARA briefings were observed for the following jobs: C steam generator lift and de-watering, spent fuel cask basket transfer, and pipe-end decontamination activities following a small release of contamination. Contract HP technician education and training were evaluated. The inspectors looked at calibration records for the air sampler monitoring the containment breach (SIC-715 attachment 1, "Portable Air Sampling Equipment Calibration Record"). Shielding calculations for a last-minute design change to the old steam generator storage facility were examined (HNP-C/SHLD-1004, Old Steam Generator Storage Facility Shielding Design).

Several AR's related to steam generator replacement activities were reviewed for indication of emerging trends and adequacy of licensee response. The AR's evaluated were nos: 48698, 48792, 48884, and 49850.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

The inspectors reviewed licensee procedure REG-NGGC-0009, “NRC Performance Indicators,” Revision 0, and various records, to determine whether submitted PI statistics had been calculated in accordance with the guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline.” These reviews were completed in accordance with inspection procedure 71151.

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors verified the PIs listed in the table below:

Cornerstone: Mitigating Systems		
Performance Indicator	*Verification Period*	*Records Reviewed*
Emergency AC Power System Unavailability	4th quarter, 2000, through 3rd quarter, 2001	Operator logs, testing records, Maintenance Rule Event logs, Equipment Inoperable records, clearance records
Residual Heat Removal System Unavailability	4th quarter, 2000, through 3rd quarter, 2001	Operator logs, testing records, Maintenance Rule Event logs, Equipment Inoperable records, clearance records

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety

a. Inspection Scope

For the cornerstone area of Occupational Radiation Safety, the inspectors interviewed cognizant personnel, reviewed and evaluated shift logs and ARs written between March 1, 2001 and October 1, 2001 to support the PI verification. ARs nos 48852, 48467, 48480, 48699, 48817, 48850, 48852, 48893, 49011, 50027, 50143, and 50200 were reviewed for assignment of responsibility, licensee evaluation, timely closure and applicability for PI reporting screening criteria.

b. Findings

No findings of significance were identified.

.3 Public Radiation Safety

a. Inspection Scope

For the cornerstone area of Public Radiation Safety the inspectors interviewed cognizant personnel and evaluated AR's written between March 6, 2001 and October 1, 2001, to support the PI verification. ARs nos 48686, 46794 were reviewed for assignment of responsibility, licensee evaluation, timely closure and applicability for PI reporting screening criteria. Radiological Environmental Technical Specification/Offsite Dose Calculation Manual Effluent calculations were evaluated for effluent doses.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the three issues identified during this period, in sections 1R05 and 1R13, to determine whether any problem identification and resolution trends were associated with them.

In relation to the discovery of foreign material in the A RHR pump containment sump suction piping discussed in section 1R13.2 and 4OA3, the following items were reviewed:

- Significant adverse condition investigation (root cause) for AR 49404, Foreign Material in RHR suction piping

- Significant adverse condition investigation (root cause) for trend AR 20874, Significant trend in FME identified problems, including training packages (MEC013H, I&C-CC013H)

- Operating Experience report for SOER 95-01 on industry FME problems, including site training package FME100G, procedure MM-011, “Cleanliness and Housekeeping,” and procedure MNT-NGGC-007, “Foreign Material Exclusion Program”

- Operating Experience report for 1992 Robinson Plant safety injection FME issue, including site training package RR-92-044

- AR titles and descriptions for the last five years that were associated with the identification of foreign material.

- The inspectors considered the inadequate extent-of-condition review for AR 49404, FME in containment suction piping, identified in section 4OA3 to determine whether it constituted another example of the cross-cutting issue identified in Section 4OA2 of NRC Inspection Report 50-400/00-04.

For the fire protection issues discussed in 1R05, the inspectors reviewed the findings from the resolution of fire barrier issues documented in Inspection Reports 50-400/99-13, 50-400/00-09, 50-400/00-01, and 50-400/01-04.

b. Findings

The inspectors found that the number of FME-related problems identified by the licensee increased during the last five years due to an increased awareness from external operating experience items, and FME sensitivity training that was conducted. Consequently, the licensee’s corrective action program identified a significant adverse trend (AR 20874), due to the increased number of FME-related ARs. The inspectors concluded that the FME-related training had resulted in an improved knowledge base in relation to foreign material in plant systems, and influenced the identification and subsequent removal of foreign material in the safety injection and containment spray piping from the containment sump. Because the material in the piping had been there a significant length of time and could have adversely affected safety system functionality, the presence of that material meant that plant risk had been higher than recognized. Consequently, the licensee’s identification and removal of the foreign material significantly reduced plant risk.

The problems with the extent of condition review for AR 49404 (discussed in section 4OA3 below) were considered to be an example of developing conclusions before enough information had been gathered and adequately analyzed to fully understand the condition. Consequently, the conclusion that foreign material did not exist in the emergency service water suction line to the turbine-driven auxiliary feedwater pump was inaccurate. This is an additional example of the cross cutting issue identified in Section 4OA2 of NRC Inspection Report 50-400/00-04.

The failure of the shutdown cooling suction valve (1RH-39) to function revealed that some maintenance problems are still being event identified rather than being identified through quality control inspections or post-maintenance testing.

The failures in the fire protection area revealed continued problems in the ability to identify and resolve fire protection issues.

The inspectors concluded that collectively the inspection findings indicated that the corrective action program was having a positive impact on risk reduction, but some issues were still being identified due to events and NRC inspections.

4OA3 Event Follow-up

(Open) LER 50-400/2001-003-00 , “1A-SA RHR Suction Line Debris - Nonconforming Condition.” This item is discussed in section 1R13.2 of this report. The licensee removed the foreign material and performed an extent of condition review as part of the root cause evaluation for AR 49404 to determine whether other unidentified foreign material intrusion situations could exist. The licensee examined the following additional stagnant piping:

- Piping from the containment sump in the B train RHR suction piping was examined with no debris found.
- Piping from the containment sump for both trains of containment spray were examined. Some debris was found in the A train containment spray piping and removed; none was found in the B train. The A train containment spray debris would not have affected system operability (ESR 01-00207).

The licensee also considered piping downstream of the RHR pump, the component cooling water system, emergency service water system, and the auxiliary feedwater system. The licensee found no additional stagnant areas that had not been previously examined.

The inspectors reviewed the extent of condition review and supporting data which included:

- inspections from the Generic Letter 89-13 service water inspection program
- descriptions of all foreign material related ARs for the last five years
- significant adverse condition investigation for AR 20874 related to an adverse trend in foreign material exclusion practices from refueling outage 9.

The inspectors found that the extent of condition review supported the conclusion that other foreign material problems most likely did not exist, with one exception. The inspectors found that the stagnant area in emergency service water supply pipe to the auxiliary feedwater pumps had not been adequately inspected, contrary to what the extent of condition review indicated. The licensee subsequently used a boroscope to inspect these pipes and found foreign material in the lines to the turbine-driven auxiliary feedwater pump. The licensee removed the foreign material and issued ARs 52640, 52645, and 52718 to address this issue. The use of this pipe is not required or assumed for any FSAR chapter 15 accidents and consequently was determined not to be significant. This item remains open pending the completion of the licensee’s corrective actions.

(Closed) LER 50-400/2000-007-02 , Technical Specification violation due to inoperable Charging Safety Injection Pump." This revision was issued to change the corrective action associated with the Charging/Safety Injection Pump monitoring probes. The inspectors reviewed completed work order 00160102 to verify the corrective action was complete. No findings were identified.

(Closed) LER 50-400/2001-001-00 , "Emergency Core Cooling System Throttle Valves Nonconforming Condition." The licensee modified the system to install orifices that would allow the valves to be opened to a value greater than the containment sump screen hole size. The inspectors reviewed completed work order 00088589 and observed the post modification testing for the throttle valves performed under procedures EST-206, "[Emergency Core Cooling System] Flow Balance," Revision 12, and OST-1801, "[Emergency Core Cooling System] Throttle Valve, [Charging-Safety Injection Pump], and Check Valve Verification 18 Month Interval Defueled," Revision 21. No findings were identified.

4OA5 Other

.1 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (TI 2515/145)

a. Inspection Scope

The inspectors reviewed the licensee's activities in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," in accordance with NRC Temporary Instruction 2515/145, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (NRC Bulletin 2001-01)," dated September 9, 2001.

b. Findings

The inspectors noted that the licensee's response to the subject bulletin included a visual examination of the reactor pressure vessel head, and that the visual examination was conducted in accordance with licensee procedure EST-227, "[American Society of Mechanical Engineers] Section XI Class 1 System Pressure Test," Revision 2. This procedure requires VT-2 examinations of various pressure-retaining boundary components, including the reactor pressure vessel head. By reference to corporate procedure NDEP-0612, "VT-2 Visual Examination of Nuclear Power Plant Components," Revision 15, EST-227 also requires that personnel who perform the visual examination be qualified as an In-Service Inspection (ISI) VT-2 Examiner Level II Limited. The inspectors verified that the examiner who completed the subject visual examination was qualified.

The inspectors considered procedure EST-227 to be an approved and adequate procedure, and that examinations performed in accordance with that procedure would be able to identify the pressurized-water stress corrosion cracking phenomenon described in the bulletin, as well as other deficiencies.

The inspectors accompanied the licensee during completion of a visual examination of the reactor pressure vessel head while the head rested on a stand on the containment operating floor. The inspectors noted that the visual examination was completed by looking through a gap of approximately 4" between the vessel head and the lower edge of ductwork that encompasses the control-rod drive mechanisms (CRDMs). Through this gap, the licensee was able to visually examine all of the CRDM nozzles, except for the eight nozzles located near the center of the top of the vessel head; the configuration of the CRDM ductwork with respect to the vessel head prevented line-of-sight visual examination of the mechanical joints between those nozzles and the vessel head. Despite their inability to visually examine those eight nozzles, the licensee concluded that none were leaking, because any leakage from those nozzles would have produced indications (like boron crystals, or streaked tracks on the vessel head where borated water flowed along the head), and because no such indications were visible.

The inspectors observed that:

- The vessel head was free of any evidence of boron leakage, except in the vicinity of a conoseal leak, and that boron deposits were visible only near the affected conoseal flange (the licensee documented the leak in AR 48414).

- A faint, light-colored film was evident on several CRDM nozzles, immediately adjacent to the joint between the nozzle and the vessel head. Generally, this film extended up the nozzle a distance of less than 0.25 inches, although on some nozzles, the film extended higher. By scraping the film, the licensee obtained samples which the licensee analyzed in a laboratory. That analysis (for sample #01-2121, documented in vault file #18-10540) found that the samples included no short-lived isotopes and only relatively long-lived isotopes. The analysis results thus indicated that the film had been deposited on the nozzles a relatively long time ago.

- The licensee hypothesized that the subject film had been deposited prior to refueling outage 8, which had been completed in late 1998. In that outage, the licensee had discovered that a leaking canopy seal weld had sprayed a small amount of boron onto the vessel head, and into many of the very small crevices that exist between the nozzles and the vessel head. After repairing the weld, the licensee had cleaned the vessel head, to remove the sprayed boron. Now, the licensee suspects that their cleaning of the vessel head at that time had failed to remove some of the boron from the crevices.

- The vessel head was also free from any debris, insulation, dirt, or boron from other sources, except that the entire vessel head was covered with a thin layer of relatively fine dust.

Except for the limitation imposed by the ductwork mentioned above, the inspectors observed no items that could impede effective examination of the vessel head.

.2 Steam Generator Replacement (SGR) Inspections

Inspection report sections 4OA5.2 through 4OA5.16 document completion of inspections that were required by IP 50001, “Steam Generator Replacement Inspection,” including several that were completed in accordance with baseline inspection procedures. The table below identifies those inspections, by correlating specific IP 50001 requirements with the corresponding sections of this report.

IP 50001 Section	Inspection Scope	Section(s) of This Report
02.02.a.1	Engineering and Technical Support Inspections (compliance with 10CFR50.59)	4OA5.3
02.02.a.2	Engineering and Technical Support Inspections	4OA5.4, 1R17,1R23
02.03.e	Operating Conditions, Radiation Protection Controls, Foreign Material Exclusion, and Temporary Services	4OA5.5
02.02.b	Engineering Preparation and Implementation for the Steam Generator Replacement Project (SGRP)	4OA5.6
02.02.c, 02.03.f	Planning & Preparation for Radiation Protection Program Controls	2OS1, 2OS2
02.02.c, 02.03.f	Implementation of Radiation Protection Program Controls	2OS1, 2OS2
02.02.a	Project Management Organization and Staffing	4OA5.7
02.02.b, 02.03.a	SGRP Procedures and Documentation	4OA5.8
02.02.b, 02.03.a	Applicable Codes and Standards	4OA5.9
02.03.a	Pre-Service Baseline Examination, Eddy Current (ET) of Replacement Steam Generators	4OA5.10
02.02.b, 02.03.b	Review of SGRP Lifting and Transportation Program	4OA5.11
02.02.b, 02.03.b	Haul Route Load Test and Evaluation	4OA5.12
02.02.b, 02.03.b	Observation of SG Lifting and Movement	4OA5.13
02.03.c	Review and Walk Down on Engineering Preparation	4OA5.14
02.03.c	Interference Removal and Restoration	4OA5.15
02.03.a, 02.03.c, 02.03.d	Special Procedures for Welding and Nondestructive Examination	4OA5.16

Section 02.02.d, “Security Considerations” was addressed in sections 3PP1 and 3PP2 of NRC Inspection Report 50-400/01-04.

.3 Engineering and Technical Support Inspections (10 CFR 50.59 Reviews)

a. Inspection Scope

As required by Inspection Procedure (IP) 50001, section 02.02.a.1, the inspectors reviewed the evaluations listed in the table below, to verify that changes to the facility as described in the corresponding ESRs were reviewed and documented in accordance with 10 CFR 50.59, “Changes, Tests, and Experiments.” To complete these reviews, the inspectors used IP 71111.02 as guidance.

Evaluation Number	ESR Number	ESR Topic/Description
99-0001, Rev. 4	97-00805, Rev. 8	Steam Generator Clear Path Removal
99-0002, Rev. 3	97-00806, Rev. 4	Steam Generator Rigging and Transport
99-0015, Rev. 1	97-00813, Rev. 2	Replacement Steam Generator Insulation
99-0016, Rev. 4	97-00807, Rev. 4	Large Bore Piping Modifications
99-0173, Rev. 3	97-00810, Rev. 3	Steam Generator Vessel Supports
99-0436, Rev. 2	97-00808, Rev. 2	Small Bore Piping
99-0556, Rev. 2	97-00809, Rev. 4	Steam Generator Instrumentation Tubing Reroute
00-0439, Rev. 0	98-00537, Rev. 0	Removal of Main Feedwater Tempering Lines
00-1352, Rev. 1	98-00534, Rev. 1*	Provide Power to a Temporary Jacking Trolley to be Installed on the Containment Bridge Hoist
01-0653, Rev. 0	00-00055, Rev. 1*	Temporarily Modify Reactor Coolant Pump “A” Circuits to Support Steam Generator Replacement

* - These ESRs described temporary modifications. (All others described permanent modifications.)

b. Findings

No findings of significance were identified.

.4 Engineering and Technical Support Inspections (Plant Modifications)

a. Inspection Scope

As required by and in accordance with IP 50001 section 02.02.a.2, the inspectors selected and reviewed several modifications that were associated with the Steam Generator Replacement (SGR) project, to verify that testing plans associated with those modifications included functional testing of the safety functions that were affected by those modifications. The selected modifications included the following ESRs:

ESR Number	Title	Revision
97-00805	"SGR [Reactor Building] Modifications"	5
97-00806	"SGR Lifting and Rigging"	0
97-00807	"SGR Large Bore Piping [Main Feedwater, Main Steam, Auxiliary Feedwater"	2
97-00809	"SGR Level Instrumentation"	4
97-00810	"SGR [Reactor Coolant System] Piping and Supports"	0
98-00537	"SGR [Main Feedwater] Tempering Valve Removal"	0
99-00466	"SGR [Steam Generator] Level Set-point Changes"	0
99-00468	"[Replacement Steam Generator] Blowdown System Modifications/Setpoints and [Steam Generator Wet Layup System"	1
94-00001	"New [Steam Generator] Component Replacement"	3

The inspectors observed/reviewed selected post-modification testing associated with these modifications, to verify that the testing will maintain the plant in a safe configuration, that no unintended system interaction will occur, that SSC performance affected by the modification meets the design basis, that testing validates the basis of any modification design assumptions, and that the modification test acceptance criteria have been met. The inspectors observed all or portions of the following post-modification testing and reviewed the test records:

Safety Function(s) Tested	Test Procedure Number	Test Procedure Title
Integrated Safeguards Actuation	OST-1825	“Safety Injection: [Engineered Safeguards Feature] Response Time, Train A 18 Month Interval on a Staggered Test Basis Modes 5-6,” Revision 12.
	OST-1080	“Auxiliary Feedwater Pump 1X-SAB Full Flow Test Quarterly Interval Mode 1, 3,” Revision 15
Main Feedwater Isolation	OST-1853	“Feedwater Isolation ESF Response Time Trains A and B, 18 Month Interval, Revision 9.”
Auxiliary Feedwater Actuation on Loss of Main Feedwater	OST-1087	“Motor Driven Auxiliary Feedwater Pumps Full Flow Test,” Revision 13
Safety Injection System Performance	EST-206	“[Emergency Core Cooling System] Flow Balance,” Revision 12
Containment Cooling Performance	OST-1010	“Containment Cooling System Operability Test,” Revision 12
ECS Throttle Valve, CSIP, and Check Valve Verification	OST-1801	“[Emergency Core Cooling System] Throttle Valve, [Charging-Safety Injection Pump], and Check Valve Verification 18 Month Interval Defueled”, Revision 21
Sequencer Block Circuit and Containment Fan Cooler Testing Train B	OST-1095	“Sequencer Block Circuit and Containment Fan Cooler Testing Train B Quarterly Interval All Modes”, Revision 14

The inspectors reviewed the following AR associated with this area to determine whether the licensee identified and implemented appropriate corrective actions:

AR Number	Title/Description
50540	ESR 01-00055 Inadequate Restoration Testing

b. Findings

No findings of significance were identified.

.5 Operating Conditions, Radiation Protection Controls, Foreign Material Exclusion, and Temporary Services

a. Inspection Scope

As required by IP 50001 Section 02.03.e, throughout this inspection period, the inspectors routinely inspected the following activities as they occurred:

- Establishment of operating conditions including defueling, RCS draindown, and system isolation and safety tagging/blocking.
- Implementation of radiation protection controls.
- Implementation of controls for excluding foreign materials in the primary and secondary side of the SGs and in the related RCS openings.
- Installation, use, and removal of temporary services directly related to steam generator replacement activities.

b. Findings

No findings of significance were identified.

.6 Engineering Preparation and Implementation for the SGRP

a. Inspection Scope

The inspectors reviewed engineering preparations including: selected ESRs, calculations, analyses, drawings, and Work Package and Inspection Reports (WP&IRs) for the SGR Containment Modifications (ESR 9700805) and SGR - Steam Generator Rigging and Transport (ESR 9700806) in order to assess adequacy and completeness. The inspectors also held discussions with SGR project management to obtain a greater understanding of the entire project scope.

b. Findings

No findings of significance were identified.

.7 Project Management Organization and Staffing

a. Inspection Scope

The inspectors reviewed the SGRP organization including: controls for contractor oversight and interface, plans for identifying and resolving non-conforming conditions, and plans for implementing quality assurance requirements in order to assess adequacy. To evaluate the SGRP project management and organization, the inspectors reviewed various documents staffing reports, forecasts, and administrative procedures and conducted interviews with various personnel in differing organizations.

b. Findings

No findings of significance were identified.

.8 SGRP Procedures and Documentation

a. Inspection Scope

The inspectors reviewed the “Special Processes Manual (SPM) for Shearon Harris Nuclear Plant Steam Generator Replacement Project” which contained procedures for welding and nondestructive examination (NDE) matrices including: procurement and control of welding filler materials, welder performance qualification standards, general welding standards, nondestructive examination standards, post weld heat treatment standards, weld documentation requirements, and welding procedure specifications.

Other procedures reviewed and compared with regulatory requirements and codes that were utilized during the SGRP included:

- AP-302 Fire Protection Housekeeping and Temporary Storage
- AP-545 Containment Entries, Rev 0
- AP-006 Procedure Review and Approval, Rev 41
- CP-10 Housekeeping and Foreign Material Exclusion, Rev 0
- MMM-011 Cleanliness and Housekeeping
- MMM-020 Operation, Testing, Maintenance and Inspection of Cranes and Special Lifting Equipment, Rev 0
- MMP-002 Installation of Piping and Piping Components, Rev 10
- MNT-NGGC-004 Scaffolding Control, Rev 0
- MNT-NGGC-005 Control of Rigging and Temporary Loads, Rev 0
- MNT-NGGC-007 Foreign Material Exclusion Area Program, Rev 0
- P8-T(RA) Bechtel Welding Procedure, Rev 5
- SPP-0602T BPC XTA7000001 Temporary Procedure for Special Processes Manual (SPM) for HNP SGRP, Rev 3
- SPP-0613T BPC XTA7000001 Temporary Procedure for Piping for HNP SGRP, Rev 0
- SPP-0614T BPC XTA7000001 Temporary Procedure for Pipe Supports for HNP SGRP, Rev 0
- SPP-0627T BPC XTA7000001 Temporary Procedure for Scaffolding for HNP SGRP, Rev 0

- SPP-0628T BPC XTA7000001 Temporary Procedure for Rigging for HNP SGRP, Rev 0
- SPP-0629T BPC XTA7000001 Temporary Procedure for Inspection and Testing of Hoisting, Rigging and Transportation for HNP SGRP, Rev 0

b. Findings

No findings of significance were identified.

.9 Applicable Codes and Standards

a. Inspection Scope

The inspectors reviewed applicable sections of the Harris FSAR, SPM and various scope documents to determine that the following Code Sections and Editions were applicable for this SGRP.

- ASME Section III (Div 1) 1974 Edition with Addenda through winter 1976 code of Construction
- ASME Section III (Div 2) 1975 Edition with Addenda through winter 1975
- ASME Section V 1989 Edition
- ASME Section IX Latest edition in effect at time of welding procedure qualification
- ASME Section XI 1989 Edition, 1992 for IWE and IWL
- AWS D1.1 Structural Welding Code-Steel (1975)
- AISC 1969, 7th Edition for supplementary steel design & re-qualification, 8th Edition for section properties and 9th Edition for New Designs
- ASCE 7-95 Minimum Design Loads for Buildings and Other Structures
- ACI 318-71 Building Code Requirements for Reinforced Concrete (Re-qualification of existing designs)
- ACI 349-80 Code Requirements for Nuclear Safety-Related Concrete Structures, Appendix B, Steel Embedments (Re-qualification of existing designs)
- ACI 349-76 Code Requirements for Nuclear Safety-Related Concrete Structures, Appendix C, Special Provisions for Impulsive and Impactive Effects (Re-qualification of existing designs)
- ASME Section III (Div 1) 1986 Edition for the Replacement Steam Generators
- ASME NQA-1 Quality Assurance Requirements for Nuclear Facility Applications, Subpart 2.15, Quality Assurance Requirements for Hoisting, Rigging, and Transporting of Items for Nuclear Plants, Section 5.3.1(a), 1994 Edition
- ANSI/ASME B30.9 Slings
- ANSI/ASME B30 Safety Standards for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks, and Slings
- ANSI/ASME B30.2 Overhead and Gantry Cranes, 1996 with B30.2a-1997 Addenda

- SNT-TC-1A — Personnel Qualification and Certification in Nondestructive Testing, 1984 & 1996 Editions
- NUREG 0612 — Control of Heavy Loads at Nuclear Power Plants, 1980
- NRC Bulletin 96-02 — Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment, April 1996

b. Findings

No findings of significance were identified.

.10 Pre-Service Baseline Examination, Eddy Current (ET) of Replacement Steam Generators

a. Inspection Scope

The inspectors reviewed the baseline eddy current data and held subsequent discussions with the licensee's cognizant corporate NDE Level III engineer, to determine that indications, burnish marks, and signals were properly dispositioned.

b. Findings

No findings of significance were identified.

.11 Review of SGRP Lifting and Transportation Program

a. Inspection Scope

The inspectors reviewed the adequacy of the SGRP lifting program to verify that it was prepared in accordance with regulatory requirements, appropriate industrial codes and standards as listed in Section 4OA5.9 of this report; and that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

The inspectors examined the SGRP lifting equipment including the inside lift system (ILS), (which included the jacking trolley (JT) and hydraulic lift unit (HLU),) the steam generator runway and the outside lift system (OLS.)

The inspectors reviewed the circumstances surrounding an incident that occurred during the assembly of the JT. Specifically while lowering both the link chain assemblies from the Temporary Lifting Device (TLD) to the spreader beam assembly, situated on top of the HLU, a “Hoist Dog” disengaged, allowing its chain assembly to fall uncontrolled onto the spreader beam assembly below. The inspectors reviewed the licensee’s followup and corrective action documented in AR 00050017, Corrective Action Report (CAR) 23638-QSSF-01-007, and Nonconformance Report (NCR) 01-045 to determine whether damage occurred to the JT that would compromise the structural integrity or use of the lifting equipment, or to permanent plant equipment.

The inspectors reviewed the following SGRP lifting documents and procedures that control these activities:

- ESR- 97-00806 R/3, SGR - Steam Generator Rigging and Transport
- Procedure 23638-SC-004-P-2394-24,R/1, Load Test Outside Lift System (OLS)
- Rigging International (RI) Drawing No. 23638-SC-004-2394-212, R/1, General Arrangement Load Test OLS Plan and Elevation
- Procedure 23638-SC-004-P-2394-44, Load OSG on Transport to OSGSF and Offload
- Drawing 23638-SC-004-2394-230, Sh. 1 & 2, Handling Seam Generators Outside Containment, Plan and Elevation
- RI Inspection Checklist - OLS Runway, 10/1/01
- RI Inspection Checklist - SG Runway, 10/3/01
- RI Inspection Checklist - OLS Trolley, 10/3/01
- RI Inspection Checklist - ILS Trolley, 10/3/01
- RI Inspection Checklist - Hydraulic Lifter, 10/14/01
- RI Inspection Checklist - Transport Trailers, 10/13/01
- Procedure 23638-SC-004-P-2394-38, Installation/Removal Jacking Trolley
- Procedure 23638-SC-004-P-2394-18, Assemble/Disassemble Jacking Trolley for Static Load Test
- Procedure 23638-SC-004-P-2394-26, Static Load Test Jacking Trolley
- Drawings 23638-SC-004-JTR-448, Sheets 1 and 2, General Arrangement Static Load Test 450 M.T. Jacking Trolley
- Drawings 23638-SC-004-C-JTR-415, General Equipment, 400 Metric Ton Lower Spreader, 250 Metric Ton Upper Spreader & 1200 Ton Swivel
- RI Load Test Certification Letter, 2/3/00 (for 450 MT Jacking Trolley)

b. Findings

No findings of significance were identified.

.12 Haul Route Load Test and Evaluation

a. Inspection Scope

The inspectors reviewed the adequacy of the SGRP transport programs, procedures, work packages and load test records, to assure that they had been prepared and tested in accordance with regulatory requirements, appropriate industrial codes, and standards, as listed in Section 4OA5.9 of this report. The inspectors reviewed the prerequisites to verify that they were met prior to commencing the test. The test involved loading the two 7-line Trabosa Hydraulic Platform Trailer Modules, linked in tandem with test weights, and then driving the transporter along the route that it would take when loaded with the old and replacement SGs.

The inspectors discussed the results of the transport path load testing with SGRP engineering personnel in order to determine that, where minor instances of sagging or settling had occurred, these areas had been excavated and backfilled with appropriate material.

The inspectors also reviewed the licensee's analyses for buried piping located beneath the transport path as documented in Calculation ID: HNP-C/YSTR-003, Rev 0, Evaluation of Safety-Related Buried Utilities along the Old Steam Generator and

Replacement Steam Generator Haul Path, and HNP-C/YSTR-0002, Rev 0, Evaluation of Buried Utilities along the Old Steam Generator and Replacement Steam Generator Haul Path.

b. Findings

No findings of significance were identified.

.13 Observation of SG Lifting and Movement

a. Inspection Scope

The inspectors observed the first of the three old steam generators being lifted from the cubicle in the reactor building to the transfer cart . Activities included: the down-ending operation which placed the generator in a horizontal position and subsequent positioning of the necessary rigging equipment to allow movement of the generator toward the equipment hatch. The inspectors also observed the movement and installation of the 3rd replacement steam generator back into containment in accordance with ESR 97-00806. During these observations the inspectors performed visual inspections of the OLS, ILS, Transfer Cart and the Temporary Lifting Device. For the task of rigging and movement of the SGs, the inspectors reviewed the ESRs for content, technical adequacy and to verify that appropriate line items had been signed off and that required pre-lift equipment inspections had been performed and documented in the enclosures provided. This review was also to verify that Industry Experience was utilized and reflected in the procedures.

b. Findings

No findings of significance were identified.

.14 Review and Walk Down on Engineering Preparation

a. Inspection Scope

The inspectors reviewed the installation of temporary pipe restraints; modification of the existing restraints; removal of snubbers, beams, and instrument lines; and pipe cuts in order to verify that the engineering preparation for the removal of SGs was in accordance with the work packages and drawings for the SGRP.

The inspectors discussed the restraint systems to be installed or modified for removal and installation of the SGs with the licensee's engineers.

The inspectors performed a walk-through inspection of the containment building, to observe the cut reactor coolant piping from the SG nozzles and observe housekeeping conditions around the work area. The inspectors looked at the corrective actions for problems identified early in the outage in the area of cleanliness, housekeeping, and control of materials and tools around the work area to verify that the problems were corrected.

b. Findings

No findings of significance were identified.

.15 Interference Removal and Restoration

a. Inspection Scope

The inspectors observed the vicinity of the all three SG cavities before the lifting operation began to make sure that the licensee had removed all the interferences and restraints. The inspectors reviewed procedures which controlled the removal and re-installation of interferences. Provisions for the temporary storage of removed interference items were also reviewed. In addition, the inspectors observed portions of the removal of interferences including piping, steam generator restraints, snubbers, and lateral supports.

After the installation of the RSGs, the inspectors observed the re-installation of various items including (but not limited to); piping, steam generator restraints, snubbers, lateral supports, and instrumentation tubing to assure that they had been installed per the engineering drawings and procedures.

b. Findings

No findings of significance were identified.

.16 Special Procedures for Welding and Nondestructive Examination

a. Inspection Scope

Reactor Coolant System (RCS) and Containment Fit Up and Welding

The inspectors conducted inspections of the fit up and welding activities involving the RCS piping and the containment equipment hatch barrel. Activities were compared with appropriate Codes and Standards as listed in Section 4OA5.9 of this report and the Bechtel Special Process Manual as discussed in Section 4OA5.8 of this report.

The inspectors reviewed the as-built configuration and held discussions with cognizant engineering personnel. This inspection was to verify that the amount of movement for the as-built "gap" associated with the cutting and fit up of the RCS piping for all three SG was within specified allowable tolerance requirements and met applicable codes.

The inspectors observed the automatic welding of the RCS hot-leg and cold-leg piping connections to the A replacement steam generator nozzles via video monitor. The inspectors observed the welding to verify that the operator at the weld and the operator at the control panel were in constant communication and to verify that the welding machine settings were being maintained within the qualified welding parameters listed in the welding procedure specification.

Training and Qualification

The inspectors, observed work, examined selected records and reviewed procedures, as listed in Section 4OA5.8 of this report to evaluate the licensee's training and qualification efforts for personnel performing cutting, machining, welding and NDE. The inspectors also reviewed the programs and compared them with the regulatory requirements and codes that were utilized during the SGRP as listed in Section 4OA5.9 of this report .

Nondestructive Examination & Post Weld Heat Treatment

As required by ASME Code Section XI, Sub Article IWA-2200, welds which have met the requirements of the applicable construction code, underwent a preservice NDE examination. The inspectors evaluated the licensee's welding, NDE and Post Weld Heat Treatment (PWHT) activities related to the SGR by conducting an inspection of the records for calibration, weld examination results, fit-up, welding, certifications of personnel and materials, and NDE (including review of radiographs.)

To verify that the radiographs showed the welds were free of rejectable indications, the inspectors reviewed the radiographs of a completed feedwater weld to verify proper penetrameter type, size, placement, and sensitivity as well as film density, identification, quality, and weld coverage. The weld selected for this work effort was as follows:

Weld No.	Size	ISO Drawing No.
1-FW-FW-4	16" x .844" (@elbow) 16" x 1.129" (@pipe)	SK-9700807-M2000

Records reviewed included WP&IRs, Field Welding Check Lists, Filler Material Withdrawal Authorizations, welding filler material Certified Material Test Reports, NDE Reports (PT, MT, and RT including radiographs), PT consumables certifications, QC inspectors and NDE examiner certification and visual acuity documentation, and certification of visual acuity examiner's qualification. Records were reviewed for completeness, accuracy and technical adequacy. The radiographs were examined for both film quality and acceptability. The following records/documents were reviewed:

NDE Examiner/QC Inspector Qualification Certification and Visual Acuity Records Examined

Examiner	Method-Level
JA	RT-II, MT, PT, VBLT
JDW	RT-II, MT, PT, VBLT
EGB	MT, PT, RTT-III, UT, VT
ANT	RT-II, MT, PT
MDD	RT-II, MT, PT, VBLT
WH	UT-III

NDE Examiner/QC Inspector Qualification Certification and Visual Acuity Records Examined

Examiner	Method-Level
JWB	UT-II
GAM	UT-II

Welder/Welding Operator Qualification Records

Welder/Welding Operator Symbol
M-877, M-820, M-872, M-1005, M-1000, M-842, M-204, 869, M-1004

Records of
Welding Filler Materials Examined

Type	Size	Heat/Lot/Batch No.
E-8018-B2 (covered)	1/8"	R016
E-805-B2 (bare)	1/8"	W109/W110, R015, R016, H109

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Scarola, and other members of licensee management at the conclusion of the inspection on January 7, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. Proprietary information was reviewed but none is included in this report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Alexander, Nuclear Assessment Manager
G. Attarian, Harris Engineering Support Services Manager
C. Burton, Director Site Operations
R. Duncan, Harris Plant General Manager
J. Eads, Emergency Preparedness Supervisor
R. Field, Regulatory Affairs Manager
W. Flanagan, SGR Project Manager
T. Hobbs, Operations Manager
J. Holt, Major Projects Manager
M. Munroe, Training Manager
T. Natale, Outage and Scheduling Manager
J. Scarola, Harris Plant Vice President
P. Summers, Environmental & Radiation Control Manager
B. Waldrep, Maintenance Manager

NRC

B. Bonser, Chief, Reactor Projects Branch 4

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-400/01-05-02	URI	Foreign Material in A RHR Containment Sump Suction Piping (Section 1R13.2)
50-400/01-05-03	URI	Failure to properly terminate a lug in the control circuit of motor-operated valve 1RH-39 (Section 1R13.3)

Opened and Closed

50-400/01-05-01	NCV	Two examples of Failure To Implement The Fire Protection Program In B Cable Spreading Room Tunnel (Section 1R05)

Closed

50-400/00-04-01	FIN	Inaccurate Risk Assessment of Startup Transformer (Section 1R13.4)
50-400/2000-007-02	LER	Technical Specification violation due to inoperable Charging Safety Injection Pump (Section 4OA3)
50-400/2001-001-00	LER	Emergency Core Cooling System Throttle Valves Nonconforming Condition (Section 4OA3)

Previous Items Discussed

50-400/2001-003-00	LER	1A-SA Residual Heat Removal (RHR) Suction Line Debris - Nonconforming Condition (Section 4OA3)