U.S. NUCLEAR REGULATORY COMMISSION
REGION I

License No.	DPR-18
Report No.	50-244/97-11
Docket No.	50-244
Licensee:	Rochester Gas and Electric Corporation (RG&E)
Facility Name:	R. E. Ginna Nuclear Power Plant
Location:	1503 Lake Road Ontario, New York 14519
Inspection Period:	October 14 through November 16, 1997
Inspectors:	P. D. Drysdale, Senior Resident Inspector C. C. Osterholtz, Resident Inspector A. Lohmeier, Sr. Reactor Engineer R. C. Ragland, Radiation Specialist
Approved by:	L. T. Doerflein, Chief Projects Branch 1 Division of Reactor Projects

EXECUTIVE SUMMARY

R. E. Ginna Nuclear Power Plant
NRC Inspection Report 50-244/97-11

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 5-week period of resident inspection. In addition, it includes the results of announced inspections by a regional specialist in inservice inspection programs, and a regional specialist in radiological exposure control programs.

Operations

Apparent human performance problems led to several system configuration control issues. None of the instances noted resulted in significant safety consequences; however, these problems adversely affected outage work controls. The issues were being addressed by operations and other interfacing groups through the licensee's corrective action process. This has been an ongoing inspector follow-up item **(IFI 50-244/97-10-01).**

Control room operators accurately performed and verified an adequate shutdown margin upon entering MODE 3. The procedure used to calculate shutdown margin appropriately incorporated the requirements specified in the Improved Technical Specifications (ITS).

The individuals who performed duties as refueling senior reactor operators (SROs) fully met NRC requirements for standing their watches, by virtue of having an active SRO license, or by properly reactivating an inactive one.

The licensee made a four hour 10 CFR 50.72 report following an automatic isolation of the containment ventilation system during removal of a steam generator manhole insert in containment. The isolation occurred as a result of radiation monitor R-12 (containment radiation) reaching its alarm setpoint. Radiation levels in the reactor coolant system had escalated during the past operating cycle due to a defective fuel assembly, and the setpoint for R-12 had been set relatively low.

The licensee made a four hour 10 CFR 50.72 report following an inadvertent automatic safety injection (SI) actuation signal on low pressurizer pressure that was generated while performing a plant safeguards system logic test. The signal was generated due to the presence of a failed pressurizer pressure bistable in combination with the test injection switch for a pressurizer pressure transmitter being in test.

Maintenance

Maintenance activities observed by the inspectors were performed in accordance with procedure requirements. Plant equipment received adequate post-maintenance testing prior to its return to service. Good personnel and plant safety practices were observed. The as-found and as-left test data met the expected performance values and the specified acceptance criteria stated in the Updated Final Safety Analysis Report.

Executive Summary (cont'd)

The licensee discovered and repaired an apparent deficiency in the B-emergency diesel generator (B-EDG) output breaker to safeguards bus 16. The licensee's intention to amend maintenance procedures to routinely check for the deficiency was appropriate.

The procedure for overspeed testing the TDAFW pump was inadequate in that it did not provide specific guidance on the operation of the trip throttle valve to ensure that the turbine would not inadvertently trip on overspeed. The licensee's intention to revise the procedure to provide more specific instructions was appropriate.

Four refueling procedures reviewed adequately provided the appropriate methods and instructions to accomplish the tasks described. No discrepancies were noted.

Better communication and attention to detail could have prevented several work control problems. Work control problems were partly related to weak configuration controls.

The guidelines for hanging test tags did not provide sufficient guidance to ensure that the tags were installed and used consistently by plant personnel. However, the licensee was quickly resolved the problem once it was brought to their attention, and the changes made to the test tag procedure were adequate to clarify programmatic requirements.

The licensee identified potential concerns regarding foreign material exclusion (FME) controls. However, the number of instances where foreign material were found were unexpectedly high. FME controls will be an inspector follow-up item to evaluate the licensee's corrective actions in this area **(IFI 50-244/97-11-01)**.

The completion status of the licensee's first refuel outage inspection in the third period of the third ten year inspection interval was consistent with the requirements of American Society of Mechanical Engineers (ASME) Code Section XI, IWB/C/D-2400 and the rules of 10 CFR 50.55a.

The licensee examination of selected components, and a safety injection (SI) Class 2 pipe line-to-elbow weld, was consistent with ASME Section XI Rules for Inservice Inspection of Nuclear Power Plant Components and Piping and 10 CFR 50.55a.

The replacement steam generator (SG) design included many features that addressed mitigation of the degradation in the original SGs. These include changes in tube material and heat treatment, and a tube support concept that should preclude collection of corrosive material. There were indications of fabrication difficulties that remain to be resolved, such as over-expansion of the tubes and inability to control the proximity of tubes to each other.

RG&E's procedures for acquisition, analysis, and evaluation of SG tube eddy current examinations were commensurate with the state-of-the-art. The data acquisition, analysis, and interpretation was well-organized and efficiently implemented. The examination program was comprehensive and exceeded the requirements of 10 CFR 50 for SG tube inspection. Examination results showed no reportable degradation. However, there were

fabrication concerns of tube over-expansion, manufacturing buff marks, and tubes in close proximity that the licensee was monitoring closely.

RG&E supported a comprehensive chemistry program that monitored all significant elements and combinations whose control is necessary for an extended plant life.

Engineering

The licensee's analysis determined that foreign material in the reactor core was the root cause of damage found in the cladding of three failed fuel pins. The foreign material was a machined stainless steel remnant, and was probably produced during reactor coolant system piping weld preparation when the steam generators were replaced in the 1996 refueling outage. The failed fuel assembly could not be reconstituted or reused in the core, and the licensee revised the core reload pattern for cycle #27 to accommodate four spent assemblies.

Modifications to motor-operated valves MOV-4007 and MOV-4008 were adequate to demonstrate reliable valve operation and adequate thrust margin for design basis conditions. The unresolved item to review the adequacy of these modifications was closed **(URI 50-244/96-06-03).**

The root cause analysis and corrective actions following the failure of motor-operated valve MOV-4007 after the 1995 refueling outage were adequate to demonstrate valve capability under design basis conditions. The unresolved item to review the analysis and corrective actions for this failure was closed **(URI 50-244/95-08-01).**

Plant Support

The licensee continued to maintain an effective radiological controls program. Radiological controls for outage work were effective in minimizing radiation dose and controlling contamination levels. The threshold for entering radiological control deficiencies into the ACTION Reporting system had decreased, and the system was effectively used to identify and resolve radiological control deficiencies. A weakness in radiological control briefings was identified, and although significant efforts were made, management has not been fully successful in improving human performance with regard to work in and around radiological boundaries.

An inspector follow-up item (IFI) was opened to review the quality and content of radiological briefings provided to plant personnel **(IFI 50-244/97-11-02).**

An inspector follow-up item (IFI) was opened to review the licensee's evaluation and use of isotopic scaling factors for internal dose assessment **(IFI 50-244/97-11-03).**

An inspector follow-up item was opened to evaluate the licensee's actions taken to investigate and document the refuel cavity/fuel transfer canal leakage pathway, and the safety significance of this condition **(IFI 50-244/97-11-04).**

Executive Summary (cont'd)

As a result of the licensee's failure to post a high radiation area around the gas decay tank (GDT), corrective actions were proposed to develop a procedural requirement for the operations group to communicate/coordinate with the health physics organization prior to degassing of the volume control tank to the GDT. Other operational evolutions that have the potential for changing radiological conditions will also be evaluated. This licensee identified and corrected item was treated as a non-cited violation (NCV 50-244/97-11-05).

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ATTACHMENTS

Report Details

I. Operations

O1 Conduct of Operations[1]

O1.1 General Comments (Inspection Procedure (IP) 71707)

The inspectors observed plant operations to verify that the facility was operated safely and in accordance with licensee procedures and regulatory requirements. This review included tours of the accessible areas of the facility, verification of engineered safeguards features (ESF) system operability, verification of proper control room and shift staffing, verification that the plant was operated in conformance with the improved technical specifications (ITS) and appropriate action statements for out-of-service equipment were implemented, and verification that logs and records accurately identified equipment status or deficiencies.

O1.2 Summary of Plant Status

At the beginning of the inspection period with the plant at approximately 95% power, the licensee was performing plant coastdown operations in preparation for the impending refueling outage. On October 16, 1997, the licensee completed a temporary modification to the main feedwater bypass piping that allowed the main feedwater bypass valves to be available for plant shutdown (see IR 50-244/97-10). At approximately 6:00 a.m. on October 20, 1997, with the plant at approximately 90% power, a plant shutdown was commenced.

The plant entered MODE 3 at 12:57 p.m. on October 20, 1997. Operations personnel noted that source range channel N-32 failed to indicate upon entering the source range. Licensee personnel subsequently discovered that the N-32 detector had failed, and the detector was replaced on October 22. Mode 5 was entered on 7:20 a.m. on October 21, 1997. The primary plant was drained to mid loop conditions, and Mode 6 was entered at 5:21 p.m. on October 23, 1997.

On October 30, 1997, the licensee identified the specific fuel assembly that contained a previously suspected fuel pin leak. This was anticipated after a noted increase in reactor coolant system (RCS) activity was observed in January 1997 (see IR 50-244/97-01). However, the assembly could not be repaired and the core had to be redesigned using four fuel assemblies that had previously been through three operating cycles (see section E2.1).

On November 3, 1997, while performing inservice inspection activities, the licensee discovered an 18 inch longitudinal crack on the B-main steam (B-MS) piping along a gusset weld near the containment penetration (see section M2.2). Subsequent

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[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

investigation revealed that similar cracking was evident on the A-MS line. However, the licensee was able to complete weld repairs on both MS lines by November 13, 1997.

Refueling operations were completed on November 4, 1997, and MODE 5 was entered at 6:39 a.m. on November 6, 1997. On November 10, 1997, the RCS was successfully refilled using a vacuum fill methodology that was utilized for the first time at Ginna Station. Both reactor coolant pumps (RCPs) were started on November 15, 1997. However, both RCPs experienced low differential pressure across the number one seal on the initial start attempt, and had to be started a second time. A plant heatup was commenced and the plant entered MODE 4 at 4:11 a.m. on November 16, 1997. MODE 4 operations were being maintained at the end of the inspection period.

O4 Operator Knowledge and Performance

O4.1 (Updated) IFI 50-244/97-10-01: Weak Configuration Control

a. Inspection Scope (71707)

The inspectors reviewed plant system configuration control deficiencies during outage activities, and evaluated the licensee's response.

b. Observations and Findings

During the current refueling outage, the inspectors noted several lapses in the licensee's control of plant system configurations. For example:

- On October 20, 1997, the inspectors noted that the licensee was running two charging pumps to increase purification flow. This was contrary to the instructions contained in procedure O-2.2, "Plant Shutdown from Hot Shutdown to Cold Shutdown Conditions," which required that only one charging pump should be running to prevent exceeding low temperature over pressure protection (LTOP) limits. The licensee subsequently determined that the operation of two charging pumps while on LTOP was an acceptable practice based upon an American Society of Mechanical Engineers (ASME) code case relief (N-514). This provision was already incorporated into the plant startup procedures, but had not yet been incorporated into the plant shutdown procedures. However, the inspectors were concerned that the plant was being operated in a different configuration than that specified in the current shutdown procedure.

- On October 23, 1997, an auxiliary building ventilation isolation occurred when a high noble gas concentration caused radiation monitor R-14 (plant vent gas) to alarm and automatically terminate a release from the gas decay tank. The release was automatically terminated. The licensee subsequently discovered that the auxiliary building supply fans were not operating, as assumed in the release permit calculations. If the auxiliary building supply

fans had been running, the automatic ventilation isolation would not have occurred since the supply fans should have provided sufficient dilution for the gas decay tank release.

- There were two instances where inadvertent reactor coolant system (RCS) leakage occurred through the B-reactor coolant pump (B-RCP) seals while work was being performed on the pump. The first instance occurred while venting the seal discharge line through valve 302B (B-RCP seal injection inlet vent valve). Maintenance workers apparently were unaware that the seals had been worked on and that the pump had not been backseated (providing a flow path). The second instance occurred when maintenance workers were apparently unaware that valve 304D (seal injection inlet block valve to the B-RCP) could not be replaced with RCS sight glass level greater than 62 inches. Actual sightglass level was at 70 inches when work began, and a flow path was created when the adjacent valve piping was cut. In neither instance was a significant amount of water drained. Licensee personnel interviewed partially attributed these problems to lapses in communication between operations and maintenance.

In addition to the events described above, the inspectors reviewed the following examples of weak configuration controls identified by the licensee during the current refueling outage, and documented in ACTION Reports for corrective action:

ACTION Report	Description and Date Identified
97-1613:	Exciter cooler valve removed with associated piping with vent valve tagged open, October 21, 1997
97-1687:	Undervoltage cabinet de-energized without hold or bus cross-tied, October 27, 1997
97-1749:	Breaker for MOV-4670 found open, October 29, 1997
97-1750:	Approved work package instructed an electrician to manipulate equipment contrary to guidance in the improved technical specification (ITS) basis, October 29, 1997
97-1780:	Decon pit damper found open during refueling operations, October 31, 1997
97-1793,1794:	Unapproved instructions found in relay room, November 1, 1997
97-1800:	Two drain valves found open during B-SG fill; caused water drip on rod drive power supply cabinets, November 6, 1997
97-1873:	Loss of 5" of reactor water vessel level when inadvertent flow path occurred during testing of valve 857B, November 4, 1997
97-1894:	Hold request modified without proper approval resulted in unexpected opening of station service bus 15 electrical breakers, November 5, 1997

c. Conclusions

The inspectors concluded that the configuration control problems noted above did not result in significant safety consequences. However, it appeared to the inspectors that these human performance problems, including the interface problems between operations and other groups, warranted management attention. The inspectors were concerned with the number of instances of configuration control problems, and there effect on work controls (see section M3.2). Pending further inspection in this area, this will remain an inspector follow-up item (IFI 50-244/97-10-01).

O4.2 Shutdown Margin Verification

a. Inspection Scope (71707)

The inspectors reviewed and verified the licensee's calculation of adequate shutdown margin.

b. Observations and Findings

The inspectors reviewed procedure O-3.1, "Boron Concentration for the Xenon Free All Rods In - Most Reactive Rod Stuck Out Shutdown Margin." The procedure referenced the requirements for shutdown margin as specified in the ITS and the Core Operating Limits Report (COLR). Control room operators identified a minimum shutdown margin requirement of 540 parts per million (ppm) boron. Actual boron concentration was later verified at 626 ppm. The inspectors performed an independent shutdown margin verification which agreed with the evaluation performed by control room operators.

c. Conclusions

The inspectors concluded that control room operators accurately performed the calculations and verified an adequate shutdown margin. The procedure used to calculate shutdown margin adequately incorporated the requirements specified in the ITS and the COLR.

O5 Operator Training and Qualification

O5.1 Refueling Senior Reactor Operator Qualification

a. Inspection Scope (71707)

The inspectors reviewed personnel qualification requirements for the individuals who assumed the refueling senior reactor operator (SRO) position required during refueling operations.

b. Observations and Findings

The inspectors reviewed the license status for the four individuals that performed duties as refueling SRO. Three individuals had SRO licenses that were active. One individual had an inactive SRO license, but reactivated the license for refueling by performing a plant tour and standing an 8 hour shift with a qualified refueling SRO.

c. Conclusions

The inspectors concluded that the individuals who performed duties as refueling SROs fully met NRC requirements for standing the watch, by virtue of having an active SRO license, or by properly reactivating an inactive one.

O8 Miscellaneous Operations Issues

O8.1 Containment Ventilation Isolation - 10 CFR 50.72 Report

On October 24, 1997, an automatic isolation of containment ventilation occurred during removal of steam generator inserts in containment. The automatic isolation occurred as a result of radiation monitor R-12 (containment radiation) reaching its alarm setpoint (2.5 E4 counts per minute (cpm), approximately one per cent of the ITS limit). Radiation levels were escalated in containment due to the presence of a defective fuel assembly (see section E2.1). The licensee generated a temporary procedure change notification (TPCN) to raise the alarm setpoint to 5.0 E5 cpm for the rest of the refueling outage. The licensee also indicated that the R-12 setpoint would be analyzed for a new permanent setpoint once the refueling outage was over. The licensee also indicated that a licensee event report (LER) would be written to address the issue.

O8.2 Inadvertent Automatic Safety Injection Actuation - 10 CFR 50.72 Report

On October 31, 1997, an inadvertent automatic safety injection (SI) actuation signal on low pressurizer pressure was generated while performing a plant safeguards logic test. The B emergency diesel generator and two service water pumps started, as designed; there was no actual injection as the rest of the systems were out of service for the outage. The licensee determined that the signal was generated due to the presence of a failed pressurizer pressure bistable (PC-430 E/F) in combination with the test injection switch for pressurizer pressure transmitter P-431 being in test. This caused 2 channels (P-430, P-431) to make up the coincidence for the SI unblock function and cause the subsequent SI on low pressurizer pressure. The bistable was replaced and calibrated, and the logic tests were successfully completed. The licensee indicated that an LER would be written to address the issue.

O8.3 (Closed) VIO 96-01-01: Both PORVs Inoperable During Stroke Time Adjustment

On March 8, 1996, the licensee authorized work that physically rendered both pressurizer power-operated relief valves (PORVs) inoperable for a total of 1 hour and 37 minutes when the reactor plant was in MODE 3. A pending temporary change to procedure M-37:150, "Copes-Vulcan/Blaw-Knox Air Operated Control Valves Inspection and Refurbishment," was altered to permit maintenance on both PORVs simultaneously without an independent review of the associated 10 CFR 50.59 safety evaluation for adequacy and completeness, as required by procedure A-601.3, "Procedure Control - Temporary Changes." Once the temporary procedure change was altered, the associated safety evaluation was no longer adequate or complete since two inoperable PORVs in MODE 3 is a condition prohibited by ITS Section 3.4.11. In addition, the shift supervisor did not adequately review the temporary procedure change for its impact on plant operations, in that he authorized work to proceed that disabled both PORVs simultaneously. Inadequate implementation of a temporary change to a maintenance procedure, and poor communications and coordination between the operations and maintenance organizations, caused both PORVs to be inoperable simultaneously. The failure to properly implement the temporary procedure change process was a violation of ITS Section 5.4.1, "Procedures."

In response to this violation, the licensee acknowledged that weaknesses existed in the procedure change process and the safety review process, in that the procedure changes were made and only the end result of the PORV maintenance was evaluated.

On August 8, 1997, the licensee revised the A-601 procedure series to require that an evaluation be made of the condition of equipment during procedure usage. On October 9, 1996, the licensee revised procedure OMG-11, "Forced Outage," to incorporate lessons learned from the event. These included a requirement to continuously man the planning office during a forced outage, and to conduct regularly scheduled status meetings. The revision also included an explicit requirement to notify the shift supervisor prior to commencing work that directly affects plant operations, or the function of safety-related and ITS equipment. On March 14, 1997, the licensee revised procedure IP-SEV-1, "Preparation, Review, and Approval of Safety Reviews." The revision added a requirement that the potential impact on equipment during performance of a procedure must be accomplished when preparing a procedure change. The revision also added a requirement for the preparer of a procedure change to determine if the plant operations review committee (PORC) is needed for an operability review of effected equipment when an interdisciplinary review would provide additional analysis of integrated plant operations.

By August 16, 1996, the licensee had completed training for all plant operators on this event and on the corrective actions taken to prevent a recurrence. The inspectors considered that the procedure changes listed above were adequate to address the causes of the violation, and noted that there has not been a repeat

occurrence of a procedure change rendering safety-related equipment inoperable. Based on the above actions, this item is closed (VIO 50-244/96-01-01).

O8.4 (Closed) VIO 96-05-01: Failure to Establish Containment Refueling Integrity

On May 21, 1996, containment penetration P-2 was not adequately sealed during the movement of irradiated fuel inside containment. A contractor removed a temporary service air line from inside the penetration leaving one 3 inch diameter hole and one 1-1/2 inch diameter hole passing through its entire length from the inside to the outside of containment. This condition had apparently persisted for approximately 3 and ½ hours. Without an adequate seal, penetration P-2 did not satisfy the containment closure conditions required by ITS section 3.9.3.c for refueling operations.

Operations procedure O-15.2, "Valve Alignment for Reactor Head Lift, Core Component Movement, and Periodic Status Checks," required that containment penetration P-2 be "adequately sealed" prior to refueling, and be independently verified by two operating personnel. Station Modification procedure SM-10034-10.01, "Temporary Service Air System For SGRP," contained a requirement to maintain the temporary service air line (in penetration P-2) pressurized or its isolation valves closed during fuel movement in containment. At the time of the event, it appeared that these procedures were not adequately implemented, and that they did not contain appropriate acceptance criteria to assure that containment integrity would be maintained during the movement of irradiated fuel. Failure to properly implement these procedures was a violation of 10 CFR 50, Appendix B, Criterion V.

Upon subsequent review and personnel interviews, the licensee was not able to determine precisely when the openings were created in the penetration, but believed that it was open for less than one hour during fuel movement. The licensee considered that penetration P-2 was adequately sealed at the beginning of fuel movement, and that procedure O-15.2 was properly implemented several hours prior to the movement of fuel when two plant operators independently verified an adequate seal. Nevertheless, the licensee considered that procedure O-15.2 should be enhanced to explicitly state that the penetration flange be in place, or that the penetration be properly sealed prior to the movement of fuel. In addition, O-15.2 needed a verification that "Do Not Remove" tags are installed on both sides of the penetration prior to the movement of fuel. These procedure changes were accomplished and became effective on September 20, 1996.

The licensee acknowledged that procedure SM-10034-10.01 was not properly implemented, but also indicated that the procedure was a special one time application for the steam generator replacement project. The procedure was only used during the 1996 refueling outage and was subsequently cancelled. The licensee also conducted training for station personnel on requirements for refueling integrity and the lessons learned from this event. The inspector observed the status

of penetration P-2 inside containment during fuel movement for the 1997 refueling outage, and concluded that it was properly sealed. Based upon the above actions, this item is closed (VIO 50-244/96-05-01).

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance Activities

a. Inspection Scope (62707)

The inspectors observed portions of plant maintenance activities to verify that the correct parts and tools were utilized, the applicable industry codes and technical specification requirements were satisfied, adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and to ensure that equipment operability was verified upon completion of post maintenance testing.

b. Observations and Findings

The inspectors observed all or portions of the following work activities:

- Removal and reinstallation reactor upper internals; observed on October 26 and November 1, 1997
- Removal of spent fuel to the spent fuel pool; observed on October 27 and 30, 1997
- Reinstallation of the reactor vessel head; observed on November 4, 1997.
- Removal of the A-circulating water (A-CW) pump; observed on October 23, 1997
- Dewatering and cleaning of both CW inlet bays; observed on November 8-10, 1997
- Replacement of the A- and B-CW pump expansion joints; observed on November 8-10, 1997
- Main feedwater bypass piping repairs; observed October 15-16, 1997.
- Erosion/Corrosion examination of the component cooling water heat exchangers; observed on October 30 and November 3, 1997
- Troubleshooting and repair of source range channel N-32; observed on October 21-25, 1997.

c. Conclusions

The inspectors concluded that the observed maintenance activities were performed in accordance with procedural requirements. Equipment received adequate post-maintenance testing prior to its return to service. Good personnel and plant safety practices were observed during the maintenance work.

M1.2 General Comments on Surveillance Activities

a. Inspection Scope (61726)

The inspectors observed selected surveillance tests to verify that equipment was properly turned over to operations and tested following the refueling outage. The inspectors also verified that approved procedures were in use, procedure details were adequate, test instrumentation was properly calibrated and used, technical specifications were satisfied, testing was performed by knowledgeable personnel, and test results satisfied acceptance criteria or were properly dispositioned.

b. Observations and Findings

The inspectors observed portions of the following surveillance activities:

- M-11.5K, "Auxiliary Feedwater Pump Turbine Major Mechanical Inspection," section 5.8, "Turbine Overspeed Trip Test," observed on October 17, 1997 (see section M2.6).
- RSSP-2.1, "SI Functional Test," observed on November 12-13, 1997
- RSSP-2.2, "EDG Safeguards Alignment Test," observed on November 14, 1997 (see section M2.5).

c. Conclusions

The inspectors confirmed that the procedures used were current and properly followed. The shift supervisor properly authorized all surveillance work to proceed. The licensee confirmed the qualifications of all surveillance test personnel involved in the tests. The as-found and as-left test data met the expected performance values and the specified acceptance criteria stated in the Updated Final Safety Analysis Report.

M1.3 Inservice Inspection Program

10 Year Inservice Inspection Interval Program Review

a. Inspection Scope (73753-02.01)

The inspector reviewed the licensee's plans and schedules for the first outage inspection of the third period of the third ten year inspection interval and determined the completion status of the inspection during the current refueling outage.

b. Observations and Findings

The inspector found that the R. E. Ginna Nuclear Power Plant "Inservice Inspection Program" met the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI and Addenda. This is consistent with requirements of 10 CFR 50.55a, "Codes and Standards."

The current refueling outage inservice inspection (ISI) is the first refueling outage in the third and last period of the third ten year interval that commenced January 1, 1990. As of October 28, 1997, the licensee's inspection schedule for the outage included 623 inspection items covering Classes 1, 2, and 3 components and piping. Since these inspections had not been completed during the inspection week, no documented report was available for review. For the previous second inspection period, the licensee had reported that the percentage completion was consistent with the requirements of ASME Section XI, IWB/C/D-2400.

The outage plan identified the system and component, the examination procedure utilized, the reference isometric drawings, the ASME Section XI Class, item, category, and facility location. The inspector found that the documentation allowed components inspected to be logically identified and categorized in terms of ASME Section XI, IWB/C/D-2500 examination and pressure test requirements.

The licensee had increased the length of the operating cycle from 12 months to 18 months. As a result of this increase, the number of refueling outages (RFOs) during the last inspection period is deceased from 3 to 2. Therefore, the total number of inspection items for each RFO inspection has been increased in both the 1997 and 1999 RFOs. The inspector found no indication that the licensee would be unable to sustain the increased inspection activity and meet the requirements of ASME Section XI, IWB/C/D-2400 for meeting the required total ISI by the end of the third inspection interval.

c. Conclusions

The completion status of the licensee's first refueling outage inspection in the third period of the third ten year inspection interval was consistent with the requirements of ASME Section XI, IWB/C/D-2400 and the rules of 10 CFR 50.55a.

M1.4 Refueling Outage Inservice Inspection Results

a. Inspection Scope (73753-2.03)

The inspector reviewed documentation of several volumetric and visual inspection results for Class 1 and 2 components to determine whether the essential examination variables had been recorded, and the test procedures and results had been implemented, reviewed, and accepted by qualified personnel.

b. Observations and Findings

The inspector reviewed results of several completed inspections, including visual inspection of Class 1 reactor stud washers, penetrant tests of Class ½ pipe-to-elbow, nozzle-to-pipe, valve-to-pipe welds, and magnetic particle tests of Class 2 integral attachments, anchors, and guides. Of those items reviewed, there were no reportable findings. The documentation for each inspection was found to be comprehensive, covering a description of the essential test variables, describing the inspection findings, identifying the participants, and providing for the necessary approvals of the results.

The type of examinations were conducted in accordance with Table IWB/C-2500 for each component. Acceptance Standards were consistent with ASME Section XI IWB/C-3000. Qualifications of the non-destructive examination personnel were found consistent with those of ASME Section XI, IWA-2300, "Qualifications of Nondestructive Examination Personnel."

c. Conclusions

The licensee examinations of selected components were performed and documented in accordance with ASME Section XI Rules for Inservice Inspection of Nuclear Power Plant Components and consistent with 10 CFR 50.55a.

M1.5 Observation of Refueling Inservice Inspection Implementation

a. Inspection Scope (73753-02.03)

The inspector observed an ultrasonic test (UT) examination of a safety injection (SI) pipe line-to-elbow weld.

b. Observations and Findings

The inspector observed the UT of a 4 inch Class 2 SI pipe line-to-elbow weld previously examined by liquid penetrant test (PT) method in accordance with licensee procedure PT-102. The base metal had also been examined by UT for laminations using licensee procedure UT-202, and none were found. The UT examination of the weld was performed using procedure UT-202. The examination used 0, 45, and 60 degree pickups. The inspector observed the calibration of the UT pickups used in the UT examination, and reviewed the qualification records of the NDE personnel.

The PT was performed consistent with ASME Section XI, IWA-2220, the UT was performed consistent with ASME Section XI, IWA-2232, and PT and UT personnel qualifications were consistent with ASME Section XI, IWA-2300. Acceptance standards were consistent with ASME Section IWC-3000, "Requirements for Class 2 Components."

c. Conclusions

The licensee examination of a SI Class 2 pipe line-to-elbow weld was consistent with ASME Section XI Rules for Inservice Inspection of Nuclear Power Plant Components and Piping and 10 CFR 50.55a.

M1.6 Replacement Steam Generator Inservice Inspection

Replacement SG Fabrication and Degradation History

a. Inspection Scope (50002-02.01)

The inspector reviewed the history of the replacement steam generators (RSGs) during their fabrication, through the initial operating period beginning in the Spring of 1996. For this period, the inspector reviewed the required tube plugging, fabrication mechanical buff marks, eddy current test (ECT) results, disposition of over-expanded tubes, and tube U-bends found in close proximity to each other.

b. Observations and Findings

The two RSGs were replacements for the original Westinghouse 44 Series steam generators (SGs). The RSGs were manufactured by Babcock and Wilcox International (BWI) in Ontario, Canada, and are conceptually similar in design in that they have vertical U-tubes with primary side reactor water entering and exiting the SG tubes in the SG channel head inlet and outlet compartments. Saturated steam is generated in the SG and leaves via the steam nozzle to the turbine. Feedwater from the completed power generation steam cycle enters the SG via the feedwater inlet nozzle through distribution rings over the tube bundle. The distribution rings have improved locations of the "J-nozzles" to preclude water hammer.

However, the RSGs have distinctly different construction features to improve the longevity of RSG life as a consequence of the extended poor operational experience of the original SG design. Each RSG has 4,765 thermally treated 0.750 inches O.D. x 0.043" wall heat treated Inconel 690 tubes (produced by Vallinox, Nucleaire, France) believed to have improved corrosion resistance over that of the original Inconel 600 SG tubes. There are 8 tube supports of egg-crate design fabricated from flat 0.10" stainless steel strips in widths of 3.15", 2.562", and 1". Ten fan bars of 1.25" width support the U-bend section of the tubes. The tube supporting system allows improved flow-through of the secondary water from SG feedwater to saturated steam as it rises through the tubes to the moisture separators. The tubes, manufactured by the pilgering process are spaced on a 1.019 inch triangular pitch and are hydraulically expanded to 0.165 inches below the full depth of the tube sheet, where the transition to nominal tube dimensions begins. The pilgering process gives minor changes in the tube inner diameter.

After manufacture, the tubes were examined before shipment to BWI, where an ASME Section XI 100% bobbin coil baseline examination of each steam generator tube was performed after installation. Also, a motorized rotating probe coil (MRPC)

examination was performed to characterize mechanical buff marks and abnormal tube expansions, and to detect tubes in close proximity.

c. Conclusions

The RSG design included many features that addressed the mitigation of the degradation found in the original SG design. These included changes in tube material and heat treatment, and a tube support concept that should preclude collection of corrosive material. There were indications of fabrication difficulty that remain to be resolved, such as over-expansion of the tubes and inability to control the proximity of tubes to each other.

M1.7 Nondestructive Examination of Replacement SG Tubes

a. Inspection Scope (50002-02.03)

The inspector reviewed the RG&E procedures for retrieval, analysis, and evaluation of the RSG nondestructive examination (NDE) results using eddy current test (ECT) probes selected for the examination. The inspector examined the status of the RG&E tube examination and findings.

b. Observations and Findings

Procedures and Guidelines

The inspector reviewed RG&E guidelines "Steam Generator Eddy Current Data Analysis Guidelines." The guidelines cover the data acquisition organization and responsibilities; history of RSG tube inspection; bobbin coil data acquisition, analysis and reporting; MRPC data acquisition, analysis and reporting; computer data screening; resolution of discrepancies; and, corrective action implementation.

Data Acquisition

The inspector visited the data acquisition center located in a remote facility outside the protected area. The digital data was transmitted by optical cable to the collection, verification, and analytic facility, where it was recorded, screened, and comparatively evaluated with tube baseline inspection data to detect any change in characteristic signals. The inspector also visited a site facility with a full scale mock-up of the RSG channel head, and tube sheet used to develop and/or test automated inspection probe locators.

Examination Program Plan

The inspector reviewed RG&E's procedure ET-110, Rev 1, "Digital Eddy Current Analysis of Inconel 690 Tubing," used for the retrieval, analysis, and evaluation of the RSG nondestructive examination results using ECT probes selected for the examination. All of the tubes (4764) of each RSG were to be examined during the current refueling outage using a 0.620M/ULC bobbin coil probe in the straight and

large bend radius U-bend tube sections, and a 0.610M/ULC probe for small radius U-bend tube sections. A randomly selected 478 (10%) tubes would be examined at the hot leg tube expansion transition using a 0.115" pancake/plus point/0.080" HF motorized rotating probe coil (MRPC). Other required examinations dictated the use of special probe configurations, examination methods, and evaluation procedures. Twenty percent of all of the two lowest radius U-bends were to be examined with a 0.115" pancake/plus point MRPC U-bend probe. Seventeen A-SG tubes and four B-SG tubes were to be examined with a 0.115" pancake/plus point MRPC U-bend probe to detect the possibility of tube proximity contact in the U-bend region.

Diagnostic MRPC examination of bobbin indications were to use 0.115" pancake/plus point/0.080 HF MRPC probe, or a 3 coil axially and circumferentially sensitive probe. Based on results of special inspections, EPRI "PWR Steam Generator Examination Guidelines," Revision 5, were to be used to expand the inspection coverage.

Results of Examinations

The inspector reviewed the status of inspection results as of October 28, 1997. At the time of the inspection, the results had not shown any reportable degradation. The licensee had completed bobbin probe inspection of 1421 of 4764 tubes (29.8%) in the A-SG, and 1420 of 4764 tubes (29.8%) in the B-SG. The licensee had completed MRPC inspection of 334 of 478 tube sheet hot leg A-SG tubes (69.9%) and 332 of 478 B-SG tubes (69.5%). The licensee had completed MRPC inspection of 13 of 24, one and two inch radius U-bend A-SG tubes (54.2%) and 18 of 24 B-SG tubes (75%). At the time of the inspection, MRPC inspections of seventeen tubes were not yet completed in the A-SG U-bends, and four of four MRPC inspections in the hot leg tubes (100%) within the tube sheet of the B-SG were complete.

The results reported during the inspection period did not indicate any tube degradation. One reported indication was that of a minor dent 2.5 inches above the hot-leg tubesheet secondary face of the B-SG. This dent was called a non-qualified indication (NQI). Also reported in a previous inspection was the over-expansion of 25 tubes. The evaluation of the SG vendor was that the degree of tube over-expansion was not detrimental to the operation of the steam generators.
As a result of surface finishing polishing of tubes during fabrication, there were 747 manufacturing buff marks detected from ECT data during this inspection. The severity of these marks had not changed since the beginning of operation of the RSGs.

Generic Tube Assembly Fabrication Anomaly

Subsequent to the onsite inspection, the inspector reviewed a Babcock and Wilcox Nuclear Steam Generator Information Bulletin, DTF-61.1.00003, August 28, 1997, that could have generic design implications. Inspections during fabrication of the RSGs determined that there was less than an optimal radial gap between the outermost tube and an adjacent tube in the U-bend region. There is a possibility of

increased wear of these tubes that could compromise the integrity of the affected tubes over long periods of operation.

BWI generic studies of the effect of close tubing proximity indicate the possibility of tube-to-tube fretting, or inter-tube deposit build-up. According to the Information Bulletin, the evaluation of the condition is still on-going. The RSG vendor believes, on the basis of studies to-date that there are no immediate consequences affecting the safety of the operating steam generators.

RG&E provided for increased surveillance of the outermost tube condition by visual inspection, where possible, and through the sensory ability of eddy current testing.

c. Conclusions

RG&E's procedures for acquisition, analysis, and evaluation were commensurate with the state-of-the-art for SG tube eddy current examination. The data acquisition, analysis, and interpretation was well-organized and efficiently implemented. The examination program was comprehensive and exceeded the requirements of 10 CFR 50 for SG tube inspection. Examination results showed no reportable degradation. Fabrication concerns of tube over-expansion, manufacturing buff marks, and tubes in close proximity, were being properly monitored by the licensee.

M1.8 Steam Generator Water Chemistry Control

a. Inspection Scope (50002)

The inspector reviewed the history and program requirements for the licensee's water chemistry program.

b. Observations and Findings

The licensee discussed with the inspector the history of steam generator chemistry control during operation of the original steam generators since start-up of the plant. The licensee believed the long operating record of steam generators with reduced incidence of tube corrosion damage (in comparison to that of other plant steam generators) at Ginna was due to the proper control of secondary water chemistry over the life of the plant.

The inspector reviewed the SG chemistry monitoring charts for the period of May through August 1997, while the plant operated at 100% power. The licensee provided strip chart evidence during this period of a comprehensive program to monitor significant water chemistry variables. These variables included reactor coolant system boron versus lithium, hydrogen, chlorides, fluorides, pH, boron, oxygen, total activity % of the technical specification limit, I-131/I-133 ratio, total gas/liquid activity, cobalt and cesium activity, actual iodine-131 activity and "equivalent" iodine-131 activity, noble gas activity, and the R-19 radiation monitor. Furthermore, monitoring secondary system chemistry included SG leak rates, SG

iodine-131 equivalent, chlorides, sulfate, sodium, blowdown volume, feedwater copper and iron, condensate dissolved oxygen, and main feedwater hydrazine.

The licensee also illustrated the data in relation to acceptance levels in the monitoring charts.

c. Conclusions

RG&E supported a comprehensive chemistry program that monitored all significant elements and combinations whose control is necessary for extended SG operability.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 A- and B-Main Steam Line Cracks

a. Inspection Scope (73753)

During the current refueling outage, the licensee discovered several large cracks in the A- and B-main steam piping inside containment. The inspectors reviewed the licensee's actions to determine the extent of the cracks and their root cause(s), and to make appropriate repairs.

b. Observations and Findings

During routine ISI inspections of the B-main steam (B-MS) piping inside containment, an 18 inch longitudinal crack was discovered in the pipe at the fillet weld of an attached gusset just inside the containment penetration (No. 402). The crack was discovered through a magnetic particle test (MT) which indicated that the crack ran the entire length on one side of the gusset. Exploratory excavations were performed and the crack appeared to be approximately 5/8 inch deep. One other crack (7-1/2 inches) at another gusset weld on the same pipe was also found through a dye penetrant test (PT). The licensee removed the gussets at each location and performed ultrasonic tests (UTs) of the affected areas to determine the defect sizes and shapes. The cracks emanated from the toe of the weld into the pipe along the edge of the weld's heat-affected zone. A portion of the largest defect was removed for laboratory analysis. The defective areas were completely excavated down to defect free base metal at a maximum depth of 0.550 inches into the pipe. The MS piping at Ginna Station is 30 inches in diameter and has a 1.25 inch wall thickness. One superficial indication (<1/8 inch) was also observed on the B-MS pipe and was removed by surface blending.

The licensee expanded the ISI inspection scope to include PT examinations of similar weld areas on the A-MS pipe, where two similar cracks (10 inches and 1-1/2 inches) were discovered at gusset welds. A similar pipe and gusset configuration also existed near the containment penetrations on both main feedwater lines, but no defects were found. In addition, other pipe attachments with fillet welds were examined on the MS piping inside and outside containment.

The licensee's metallurgical analysis revealed that the MS pipe cracking was due to "lamellar tearing" that resulted from a poor weld design, and excessive weld shrinkage that caused a tensile overload in the pipe's base metal. The gusset plates were attached to the pipe in a "Tee" configuration with full penetration fillet welds. This configuration caused high residual stresses in the heat affected zone of the weld. The analysis also revealed that the defects were "self relieving" and had become dormant soon after the failures occurred. No residual stresses were present after the tearing stopped. A heavy buildup of oxide material inside the cracks indicated that the defects have probably existed since original construction. The type of oxidation was typical for high welding temperatures, and did not occur at the lower temperatures seen during normal plant operation. The previous ISI inspections at the main steam line penetrations were performed in 1989, but the cracks were not readily visible. The licensee indicated that only a visual examination was required by the ASME Code at that time. For the licensee's current third ten-year ISI interval, the code required an MT inspection, and the cracks were discovered during this outage.

The licensee removed all defective portions of the MS piping and performed weld repairs in accordance with the original pipe construction specification (B31.1). The gussets that were removed were not reinstalled since they were not credited in the plant's steam line break analysis.

c. Conclusions

The licensee thoroughly examined the MS piping to determine the number and extent of original weld defects. The examinations were expanded beyond the minimum scope of ASME Code requirements to include the main feedwater piping penetrations and other pipe attachments inside and outside containment. The licensee's laboratory analysis provided an in-depth root cause evaluation and assessment of the weld defects.

M2.2 Installation of Plant Modifications

a. Inspection Scope (62707)

The inspectors reviewed the contents of two modification installation packages for completeness, and acceptability of safety reviews, material issue sheets, and quality control hold point verifications. The inspectors also conducted field observations of the accessible portions of the installed equipment, and held discussions with the system engineers and operators to evaluate the acceptability of the modifications.

b. Observations and Findings

Reactor Protection System Channel #2 Resistance Temperature Detector (RTD) Module Replacement

This modification replaced the resistance temperature detector (RTD) input processing modules in the channel 2 (white) protection rack to reconfigure the

module scheme such that the failure of one RTD will still allow for continuous Tave and Delta-T signal processing. The previous processor utilized a resistance bridge circuit to generate the Tave and Delta-T signals. The new modification averages the RCS hot leg and cold leg RTD inputs for generating a signal to the reactor protection system. If one RTD should fail, its input can now be removed from the circuit and replaced by doubling an input from an operable RTD. The inspectors concluded that the modification was appropriate in that its installation could prevent the necessity of a plant shutdown should one RTD fail. The inspectors also reviewed the work package and associated safety evaluation (SEV-1094), and found both to be satisfactory.

Core Exit Thermocouple Panel Replacement

Prior to the current refueling outage, the licensee's core exit thermocouple (CET) display panel in the control room had become obsolete since it was no longer manufactured and spare parts were no longer available. A new display panel was procured and installed in the control room during the outage. Four CET detectors were disabled prior to the outage due to failures, and all were repaired during the panel replacement. The new panel provided an improved display that was easy to read, provided continuous CET temperature updates, displayed the minimum and maximum core temperature for both CET groups in each instrument channel, calculated the average temperatures for each channel, and provided alarm indications for system malfunctions. Operations personnel considered the new display to be a significant improvement that would simplify monitoring of incore temperatures.

c. Conclusions

Both the RPS module and CET display modifications were good enhancements to the reliability of plant equipment and operations. The installation packages reviewed contained complete information, verification signatures, and adequate safety evaluations.

M2.3 Circuit Breaker Outage Maintenance and Output Breaker to Safeguards Bus 16 Troubleshooting

a. Inspection Scope (62707)

The inspectors reviewed the scope of outage breaker maintenance and the licensee's efforts to troubleshoot a failure of the B-emergency diesel generator (EDG) output breaker to safeguards bus 16.

b. Observations and Findings

The licensee performed a significant amount of maintenance on Westinghouse breakers during the refueling outage. Included in this maintenance were breakers for the EDGs, RCPs, component cooling water pumps, main feed pumps, circulating water pumps, condensate pumps, condensate booster pumps, and heater drain

pumps. No significant deficiencies were noted during the maintenance. Several breakers had some secondary contacts replaced, as they were noted to be in the early stages of degradation. Closer inspection of secondary breaker contacts was incorporated as part of the licensee's recently completed root cause analysis on secondary contact failures (see IR 50-244/97-06).

However, on November 14, 1997, the B-EDG output breaker to bus 16 failed to shut as expected when an undervoltage condition was caused on bus 16 during the performance of test RSSP-2.2, "EDG Safeguards Alignment Test." This breaker has had a history of performance problems (see IRs 50-244/96-11 and 50-244/96-12). The breaker's relays were checked satisfactorily. RSSP-2.2 was performed a second time and the breaker shut properly, but a third test was performed with test equipment installed, and the breaker again failed to shut. Consequently, the breaker was removed and taken to the electrical shop for troubleshooting and bench testing. Electrical technicians discovered that the breaker's shunt trip coil plunger mechanism was sometimes not fully resetting due to increased mechanism resistance, and was preventing the breaker from closing. After the shunt trip coil plunger mechanism was replaced, the breaker was tested satisfactorily and returned to service. The licensee performed a visual check of other similar Westinghouse breakers and noted no discrepancies. The licensee also indicated that specific mechanical checks of the shunt trip coil plunger would be incorporated into electrical maintenance procedures.

c. Conclusions

The inspectors concluded that the licensee discovered and repaired a deficiency in the B-EDG output breaker to safeguards bus 16. The licensee's intention to amend maintenance procedures to check for the deficiency was appropriate.

M2.4 Turbine Driven Auxiliary Feedwater (TDAFW) Pump Turbine Overspeed Trip

a. Inspection Scope (62707)

The inspectors reviewed the licensee's performance of an overspeed trip test for the TDAFW pump turbine.

b. Observations and Findings

On October 17, 1997, while performing procedure M-11.5K, "Auxiliary Feedwater Pump Turbine Major Mechanical Inspection," section 5.8, "Turbine Overspeed Trip Test," the TDAFW pump turbine prematurely tripped on overspeed. M-11.5K did not specifically direct operators to ensure the trip throttle valve was closed prior to opening the steam admission valves. This test is an infrequently performed procedure, and the licensee indicated that the individual who had historically always performed the test had recently left RG&E. The licensee initiated an ACTION Report (97-1586) to address the issue, and indicated that procedure M-11.5K would be revised to provide more specific instruction on the operation of the trip throttle valve.

c. Conclusions

The inspectors considered that the procedure for overspeed testing the TDAFW pump was inadequate in that it did not provide specific guidance on the operation of the trip throttle valve to ensure that the turbine would not inadvertently trip on overspeed. The licensee intention to revise the procedure to provide more specific instructions was appropriate.

M3 Maintenance Procedures and Documentation

M3.1 Refueling Procedures

a. Inspection Scope (60705, 60710)

The inspectors reviewed selected refueling procedures for adequacy.

b. Observations and Findings

The inspectors reviewed the following procedures:

- RF-73.1, "Fuel Assembly Repair Procedure,"
- O-15.2, "Valve Alignment for Reactor Head Lift, Core Component Movement, and Periodic Status Checks,"
- RF-65, "Reactor Refueling Procedure Volume I: Refueling Procedure Instructions,"
- STD-400-162, "Procedure for PWR In-Mast Telescope - Sipping."

c. Conclusions

The inspectors concluded that the reviewed refueling procedures adequately provided the appropriate methods and instructions to accomplish the tasks described. No discrepancies were noted.

M3.2 Outage Work Controls

a. Inspection Scope (62707)

The inspectors reviewed several deficiencies in the licensee's control of outage work activities.

b. Observations and Findings

On October 26, 1997, the inspectors noted that maintenance personnel were performing work in the B-EDG room even though there were postings in the area indicating that the B-EDG was in protected status and that no work should be performed. The licensee subsequently initiated an ACTION Report (97-1679) to address the issue. Licensee personnel interviewed stated that better communications between departments could have prevented the deficiency from occurring.

The inspectors noted that the licensee had also identified a number of work control deficiencies and had initiated ACTION Reports to address these problems. Examples are listed below:

ACTION Report	Description and Date Identified
97-1603:	Premature holding of the A-RCP, October 21, 1997
97-1768:	Work almost started above the B-CCW pump while it was in protected status, October 30, 1997
97-1784:	Safety-related material tagged with a no-QA tag, October 31, 1997
97-1852:	Missed surveillance on boron concentration verification, November 3, 1997
97-1857:	Wrong authorized person on hold requests, November 4, 1997
97-1866:	Inappropriate pen and ink change made for work on the D-containment recirculation fan, November 3, 1997
97-1890:	A-SG level lowered too early (prior to visual inspection of the secondary side), November 2, 1997
97-1892:	Wrong fuses pulled during isolation of station service transformer, November 5, 1997
97-1912:	A-battery rack modification not properly turned over to operations, November 8, 1997
97-1930:	Wrong valve issued from stock, November 9, 1997
97-1933:	B-main steam line heat treatment equipment installed without approved work package, November 10, 1997
97-1936:	Plant Change Notices written in May 1996 still not through independent review, November 10, 19997

c. Conclusions

The inspectors considered that better communication and attention to detail could have prevented some of the work control problems noted above. The inspectors also considered that work control problems were partially responsible for degraded configuration controls noted in section 04.1.

M3.3 Test Tag Control Program

a. Inspection Scope (62707)

The inspectors reviewed the licensee's program for controlling test tags.

b. Observations and Findings

The inspectors reviewed procedure A-1103, "Test Tag Control Program," and noted multiple instances in the plant where test tags were not being filled out consistently. Operations personnel in general did not appear to be knowledgeable about the program or its tracking requirements. The guidance in procedure A-1103 directed personnel to "fill out pertinent information" on the tags. However, there were some cases where sections on the tags were not filled out, and other cases where more complete information was provided. These inconsistencies did not appear to be an immediate safety concern, as the minimum information (i.e. component name and required status) was included on all tags observed by the inspector. Subsequently, the licensee revised procedure A-1103 to specifically identify expectations for individuals hanging tags to fill out all information that is requested on the tag, and routed the changes to operations personnel to communicate those expectations.

c. Conclusions

The inspectors concluded that the guidelines for hanging test tags did not provide sufficient guidance to ensure that test tags contained consistent information. However, the licensee quickly resolved the problem once it was brought to their attention, and the changes made to the test tag procedure provided appropriate program requirements.

M7 Quality Assurance in Maintenance Activities

M7.1 Foreign Material Exclusion Controls

a. Inspection Scope (62707)

The inspectors reviewed the licensee's controls for foreign material exclusion (FME).

b. Observations and Findings

FME problems at the Ginna plant have occurred on several occasions during recent months (see NRC IRs 50-244/94-25, 95-20, and 97-09). In October 1997, the licensee discovered foreign material inside the suction piping of the reactor coolant drain tank pumps. As a result, ACTION Report 97-1507 was initiated to develop a plan to identify and evaluate systems that could be susceptible to foreign material intrusion, so that additional inspections could be performed on systems prior to the end of the current refueling outage. During the current outage, foreign material was

identified as the root cause of the three failed fuel pins discussed in section E2.1 below. The licensee also identified other multiple instances of FME control deficiencies during the current outage work, and incorporated them into the ACTION Report system. Examples of these deficiencies included:

ACTION Report	Description and Date Identified
97-1674:	Loose items lying around low pressure turbine, October 25, 1997
97-1722:	Lube oil cooler left uncovered, October 27, 1997
97-1800:	Wrench found in lube oil reservoir, October 30, 1997.
97-1828:	Bolt and nut found in B-steam generator, November 3, 1997
97-1875:	Plugged test connection, November 4, 1997
97-1876:	Brass shavings found in drain line, November 4, 1997
97-1881:	Metal found in service water inlet piping, November 4, 1997
97-1944:	Nut dropped on reactor head, November 11, 1997
97-1956:	Brass particles in instrument air regulator, November 13, 1997
97-1963:	Piece of metal inside steam dump valve, November 13, 1997

c. Conclusions

The inspectors concluded that the licensee appropriately evaluated potential concerns regarding FME controls, and entered specific FME deficiencies into the corrective action system. However, the inspectors were concerned that the number of instances were unexpectedly high. This will become an inspection follow-up item to evaluate the licensee's corrective actions regarding FME controls **(IFI 50-244/97-11-01).**

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) URI 96-03-01: UFSAR Pertaining to ISI/NDE Not Consistent With ASME Code Section V

During an ISI/NDE inspection in April and May 1996, the NRC reviewed applicable portions of the Ginna Updated Final Safety Analysis Report (UFSAR). An inconsistency was noted between the UFSAR and the plant practices, procedures, and/or parameters used in the ISI/NDE programs observed by the inspectors. Paragraph 5.2.4.4 referenced Appendix III and IV of the ASME Code, Section V. However, ASME Section V, 1986 Edition, did not contain these appendices. Ginna updated to the 1986 Edition of the ASME Code for the current inspection interval, as described in 10 CFR 50, Section 55a(g)(4). RG&E agreed to correct the references in Paragraph 5.2.4.4 of the UFSAR in the next revision. This item was left unresolved pending correction of the UFSAR and any ISI/NDE procedures containing these references.

The licensee revised UFSAR Section 5.2.4.4 in December 1996. The incorrect reference to Section V, Appendices III and IV of the 1986 version were deleted. The correct reference was ASME Code Section XI for this part of the UFSAR and

was incorporated into the revision. No reference to the incorrect Code section existed in any existing ISI/NDE procedures as of this inspection. Based upon the above actions, this item is closed **(URI 50-244/96-03-01).**

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Root Cause for Failed Fuel Pins and Revised Core Loading Pattern

a. Inspection Scope (37551, 60710)

The inspectors reviewed the licensee's activities to determine the root cause(s) for three failed fuel pins. The inspector also reviewed the licensee's revised fuel reload pattern and safety evaluation that was necessary after the failed assembly could not be reused in the reactor.

b. Observations and Findings

In January 1997, the licensee identified high reactor coolant system (RCS) activity that indicated failed fuel (see IR 50-244/97-01). During refueling operations and sipping of core assemblies on October 30, 1997, fuel assembly E71 showed evidence of leaking pins. The assembly was moved to the spent fuel pool, where the pins could be replaced. UT inspection of this assembly revealed four fuel pins with signs of degradation. Three pins were replaced with "dummy" pins, but only two of these pins actually had breached cladding. The fourth pin could not be completely removed because it was severed through at its end cap, and also at approximately 30 inches below the top of the pin. Two locations at lower elevations were also fully severed through the pin cladding. Underwater video inspection of the three failed pins revealed finely worn full penetration grooves at approximately the same elevation near the bottom of the pins. These three pins also exhibited through-wall circumferential wear and hydride blisters that exposed fuel pellets. During removal of the top segment of the fourth pin, no pellets were dropped or lost from the broken segments. The licensee determined that assembly E71 could not be reused in the reactor core since the entire pin could not be readily extracted without a risk of losing fuel pellets into the spent fuel pool.

The licensee determined that the root cause for the failed fuel pins was foreign material left in the RCS from the steam generator replacement project during the 1996 refueling outage. A small metal chard was discovered in the refueling cavity that was a remnant from RCS piping weld preparation for the new steam generators. The metal piece was apparently caught in a lower flow nozzle of the E71 assembly and wore through-wall groves in the nearby fuel pins. The size of this remnant closely matched the defect found in the failed fuel, and all groove defects were at approximately the same area of the assembly. The licensee concluded that the foreign material was the cause of the pin failures and led to more extensive degradation. Westinghouse was pursuing a more detailed analysis

to confirm the licensee's conclusions.

The initial core reload pattern and the purchase of new fuel assemblies for cycle #27 was based upon an assumption that the failed assembly could be reconstituted and reinstalled into the core. Since no other fuel bundle was available to replace the unusable assembly, the licensee redesigned the cycle #27 core with a new loading pattern. Four old assemblies were installed at peripheral positions, and 32 other assemblies were relocated to optimize flux and power distribution in the modified core. The licensee prepared a safety evaluation for the revised core that incorporated a revised reload safety evaluation provided by Westinghouse. The safety evaluation provided adequate justification to demonstrate that the revised core did not represent an unreviewed safety question, and that the probability or consequences of any accident described in the UFSAR would not be increased.

c. Conclusions

The licensee identified foreign material that was the most likely cause of the failed fuel pins. The revised reload pattern for the cycle #27 core was well supported by a safety evaluation.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) URI 96-06-03: Modifications to Auxiliary Feedwater Valves MOV-4007 and MOV-4008

During the 1996 refueling outage, the licensee replaced valves MOV-4007 and MOV-4008 after high internal erosion and unreliable throttling was observed during surveillance tests. Identical valves were not available, but valves thought to be hydraulically "equivalent" were substituted. However, a slightly different plug design may have caused the new valves more difficulty in achieving the proper throttle flow under varying SG pressures and valve d/p conditions. The valves were only about 5% open at the correct throttle position. An unresolved item was opened to review this modification as part of the Ginna MOV program.

While the post-modification test results of MOV-4007 and 4008 were adequate, they continued to exhibit unreliable automatic throttling during surveillance tests, and also evidenced little margin available under degraded voltage conditions to account for degradation. The valves' reliability to automatically throttle flow appeared related to an inadequate design of the valves and their controls. The licensee indicated their intention to modify the valves, actuators, and control circuits to increase their throttle reliability and output capability.

The licensee continued to investigate long term corrective actions to improve throttling reliability. The valves were again modified by installing throttling plugs when the plant was shutdown for maintenance in September 1996. Post modification testing demonstrated that the valve's control characteristics produced more predictable and reliable automatic throttle flow. However, the licensee noted anomalies in high packing loads and valve/actuator inertia (see IR 50-244/96-09).

The licensee performed a modification to correct this problem which included an actuator gear change to decrease the speed of the valve. The gear change proved sufficient for MOV-4008, but MOV-4007 could not be modified in this way. Consequently a modification was performed on its close limit switch to fully close the valve without jeopardizing the total torque limit. It was not a safety-related function to fully close these valves; however, full closure capability was more desirable from an operational standpoint. The licensee planned to reconfigure the limit switch on MOV-4008 to make it close on stem position instead of motor torque during a future outage. This was accomplished in a subsequent outage in November 1996, when the plant was shutdown to replace the station's main output transformer. Following that outage both MOVs demonstrated repeatable and reliable automatic throttling. MOV-4007 had 16% margin and MOV-4008 had 23% margin under degraded voltage conditions.

The licensee continued to evaluate these valves and to provide long term corrective actions to an apparently inadequate design. Options were evaluated which included replacement of the valves with ones having better control characteristics, installation of a more optimum actuator gear ratio, and improvements in the valve control circuitry. During the 1997 refueling outage, the licensee replaced the actuator on MOV-4008 with a larger Limitorque model SMB-00. The actuator provided a higher thrust capability and a larger thrust margin for design basis conditions. MOV-4008 was dynamically tested under differential pressure conditions and demonstrated reliable proper throttling capability. MOV-4007 was also tested satisfactorily through a normal surveillance test. The MOV-4007 actuator was not replaced due to insufficient time in the outage and a potential fitup problem. Consequently, replacement of its actuator was deferred to the 1999 refueling outage. However, the tested reliability of these valves to throttle properly upon automatic actuation, and the improvement in thrust margins demonstrated as they are currently set up are sufficient to demonstrate that the modifications are adequate to assure the valves will perform as expected under design basis conditions. This item is closed **(URI 50-244/96-06-03).**

E8.2 <u>(Closed) URI 95-08-01: Evaluate the Root Cause Analysis and Corrective Actions for MOV-4007 Failure</u>

Auxiliary feedwater motor-operated valve MOV-4007 failed to open during a monthly surveillance test in April 1995. The valve did not open on a demand signal from the control room and its motor overheated and failed because the valve disk was locked into its seat. The motor did not trip off on high current because its thermal overload protection did not function. The valve locked into its seat during a previous test when the valve was going closed under full inertia and maximum motor torque. The test was performed at maximum differential pressure with the valve disk at its required throttle position. Very high vibration at the valve and adjacent piping was seen during the test that apparently loosened the actuator torque switch set screw and caused the torque switch to change to its maximum setting. At the maximum torque switch setting, the motor achieved a torque output that was higher (157%) than the maximum torque rating for the actuator. A limiter plate was not installed on the torque switch to prevent the motor from exceeding

the actuator's maximum rated torque. The unresolved item was opened to evaluate the licensee's root cause and corrective actions for this failure.

After the failure, the licensee disassembled the valve and actuator, replaced the motor and other damaged components. High vibration had never been observed in MOV-4007, but the licensee concluded that it was the result of the high differential pressure (d/p) across the valve at its normal throttle position, since the d/p was the maximum that could be seen by the valve under design basis conditions. The licensee also reevaluated the MOV program policy on the use of limiter plates, and revised the MOV program documents to clarify the requirements for limiter plates. The appropriate limiter plates were installed on MOV-4007 and MOV-4008 that were suitable to the current actuators and their maximum torque and thrust limitations.

The licensee concluded that the torque switch set screw was not properly secured following a previous adjustment and that the high vibration caused the torque switch to loosen. Consequently, the licensee revised the procedures with instructions for making torque switch adjustments to include additional detail for properly securing the set screws. After several valve modifications, and installation of the new steam generators, multiple tests on both valves at the maximum design basis differential pressure have been performed with no observed recurrence of high vibration. As discussed in section E8.1 above, repeatable tests for both valves have demonstrated reliable and repeatable throttling capability. Based on all of the above actions, this item is closed **(URI 50-244/95-08-01).**

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Refuel Outage Radiological Controls

a. Inspection Scope (83750)

A review was performed of radiological controls implemented for outage work. Information was gathered by a review of radiation exposures, contamination controls, control of radiological boundaries, radiological briefings, and use of isotopic scaling factors.

b. Observations and Findings

Radiological controls for outage work were effective in minimizing radiation dose and controlling contamination levels. Projected total radiation dose for the refueling outage was estimated to be less than 60 person-rem; contamination levels in the containment building were maintained less than 2,000 disintegrations per minute (dpm)/100 cm^2 in general areas and less than 10,000 dpm/100 cm^2 in high contamination areas; and controls were effective in minimizing internal exposures in spite of increases in alpha contamination levels in containment associated with the

discovery of four breached fuel pins. Health physics staff members were conscientious in efforts to monitor radiological boundaries and inform plant workers of radiation sources and low dose areas.

Radiological Briefings

Radiological briefings provided to personnel for work in the containment building were generally good. Health physics technicians ensured that personnel sufficiently described job scope, and that personnel were familiar with radiological conditions and radiological controls for described work. However, a weakness with respect to radiological briefings was identified. Procedure A-1.8, Radiation Work Permit, Rev. 3, step 3.2.3 stated that (in response to alarming dosimetry) "workers should leave the area and notify RP (radiological protection) personnel if RADOS is alarming." Contrary to this procedural requirement, two health physics technicians informed the inspector during a prejob briefing that it was not necessary to notify health physics of a RADOS alarm, if the alarm was only a "dose-rate" alarm and not a "dose" alarm. The technicians stated that this was acceptable as long as the inspector immediately backed out of the area and the audible alarm stopped (indicating that the alarm was only due to dose-rate).

The inspector raised the concern to health physics management that the instruction did not appear to be fully consistent with procedural guidance, required non-health physics personnel to distinguish between dose and dose rate alarms, and was not conservative. The health physics manager performed a review of the issue, and stated that the general policy was for all personnel to exit the work area and contact health physics upon initiation of any RADOS alarm. However, the technicians performing the radiological briefing were familiar with radiological conditions in containment, knew that the preset dose-rate alarm on the inspector's RADOS could go off in several specific areas, and extended greater autonomy to the NRC inspector than would be provided to general station workers. An inspector follow-up item (IFI) was opened to review the quality and content of radiological briefings provided to plant personnel **(IFI 50-244/97-11-02).**

Isotopic Scaling Factors

The inspector reviewed licensee efforts to determine if isotopic scaling factors had changed in light of the discovery that four fuel pins were found to be damaged. Smears were obtained in areas expected to represent primary water including the bottom of the reactor head and the cono-seals. A composite sample was made, the sample was analyzed, and the sample was sent to a vendor for isotopic analysis. Results of the vendor analysis indicated that alpha contamination levels had increased. Cesium-144 (Ce-144) activity (an indicator of transuranics) had increased with respect to Co-60 by a factor of approximately two, and transuranic activity had increased by a factor of about 2.5 with respect to Ce-144. The principle health physicist indicated that air sample and whole body count data would continue to be carefully reviewed for the presence of Ce-144 and transuranics, that additional samples would be taken to establish current isotopic ratios (scaling factors) to assess internal dose, and that some changes to software used to

calculate internal dose would need to be revised. However, Ce-144 had not been detected in any whole body counts performed during the outage, and no significant internal doses requiring internal dose assignment had been received for the outage. An inspector follow-up item (IFI) was opened to review the licensee's evaluation and use of isotopic scaling factors for internal dose assessment **(IFI 50-244/97-11-03).**

c. Conclusion

Based on this review, the inspector concluded the following:

- Radiological controls for outage work were effective in minimizing radiation dose and controlling contamination levels.
- Information provided during a radiological control prejob briefing was not fully consistent with procedural guidance; accordingly, an inspector follow up item was opened to evaluate the content and quality of radiological briefings.
- Results of a vendor analysis of a composite smear sample indicated that alpha contamination levels had increased. An inspector follow up item (IFI) was opened to review the licensee's evaluation and use of isotopic scaling factors for internal dose assessment.

R2 Status of Facilities and Equipment

R2.1 Housekeeping and Material Conditions

a. Inspection Scope (83750)

The inspector performed a review of housekeeping and material conditions. Information was gathered through plant tours and discussions with cognizant personnel.

b. Observations and Findings

Housekeeping in the primary auxiliary and intermediate buildings was generally good in that walkways and aisles were clear and free of debris, and work areas were well illuminated. Housekeeping deficiencies were identified in the containment building in that aisles and walkways and work areas were cluttered with multiple bags, hoses, cables, equipment, and some trash. The inspector noted that an ACTION Report (97-1877) had been initiated by the quality assurance organization relative to housekeeping in the containment building.

The inspector also noted that walls adjacent to the refuel cavity/fuel transfer canal within containment leaked while the cavity was flooded, resulting in an accumulation of water and puddles on the lower elevation of containment. Based

on a review of sump water level increases, the resident inspector estimated that the total leak rate for this pathway was approximately 4 gallons per minute. To avoid accumulation of water on the containment floor, work crews were sent into containment approximately every hour to squeegee and direct water to floor drains. the licensee stated that the condition had existed for an extended time, and was previously evaluated. However, no repairs were warranted due to the perceived high cost and low safety significance. An inspector follow-up item was opened to evaluate the licensee's actions taken to investigate and document the refuel cavity/fuel transfer canal leakage pathway, and the safety significance of the condition (IFI 50-244/97-11-04).

c. Conclusion

Based on this review, the inspector concluded that housekeeping was generally good in the primary auxiliary and intermediate buildings, and generally poor in the containment building. In addition, the material condition of the refuel cavity wall was somewhat degraded in that water leaked from the refuel cavity to the containment floor.

R5 Staff Training and Qualification in RP&C

R5.1 Staff Training

a. Inspection Scope (83750)

The inspector reviewed the content of a special training session provided to station and contractor health physics technicians that incorporated details of a recent industry event. Information was gathered by a review of training materials and discussions with a training instructor.

b. Observations and Findings

Procedure training was conducted for plant and contract health physics technicians that included a recent industry event involving a failure of health physics personnel to fully assess internal dose due to transuranics, breakdowns in communications, and a failure of contamination monitoring equipment. The training session was presented as a potential outage scenario using Ginna radiation work permits (RWPs), radiological survey forms, and air sample data sheets. Diagrams of plant areas, radiological survey instruments, and audio recordings of radiological briefings were used to add realism, and frequent class discussions were included. Training materials were well detailed, were logically presented, and were challenging.

c. Conclusions

Based on this review, the inspector concluded the quality and content of a special training session provided to station and contractor health physics technicians, that incorporated details of a recent industry event, were excellent.

R7 Quality Assurance in RP&C Activities

R7.1 Use of the ACTION Report System for RP&C Issues

a. Inspection Scope (83750)

The inspector performed a review of the effectiveness of the ACTION Reporting system for identifying and resolving radiological control deficiencies. Information was gathered by a review of approximately 40 ACTION Reports written to address deficiencies in radiological controls, and through discussions with cognizant personnel.

b. Observations and Findings

Approximately 40 ACTION Reports (ARs) related to radiological controls were written during the first two weeks of the refuel outage. Many of the deficiencies were corrected on-the-spot, but were entered into the ACTION reporting system for trending purposes (priority D ACTION Reports). Examples included AR 97-1772, health physics administrative delays; AR 97-1650, worker on wrong radiation work permit (RWP); and AR 97-1600, workers not wearing electronic dosimetry in the correct position. The inspector noted that reports of ARs could be readily generated and included AR number, event date, subject, description, cause code, priority code, and responsible department.

Equipment Hatch

During previous refuel outages, equipment was transported into and out of containment by removing the equipment hatch and moving cargo vans up to the containment equipment hatch penetration for loading/unloading. An enclosure (dog house) at the equipment hatch served to protect the equipment hatch from elements of weather. In response to SECY 97-168, the licensee left the equipment hatch (with a personnel door) in place, to allow the containment penetration to be readily closed during shutdown operation. The equipment hatch personnel door extended well beyond the equipment hatch penetration and was not fully protected from elements of weather by the equipment hatch "dog house." Licensee staff recognized that potentially contaminated equipment could be exposed to elements of weather during equipment loading/unloading and initiated AR 97-1823. To address this concern, a scaffolding structure with tarpaulins was erected to act as a weather shield. The resident inspectors examined this weather shield during the week of November 10, 1997, and found that it provided adequate weather protection.

Unposted High Radiation Area

One AR (97-1597) was written to address the creation of an unposted high radiation area, and was selected for further review. On October 20, 1997, the volume control tank (VCT) was vented to the A-gas decay tank (A-GDT) ahead of schedule and without coordination between the operations group and the health physics staff. Due to increased noble gas activity in the primary system (associated with known fuel damage), this transfer created a high radiation area with dose rates as high as 350 mR/hr in the vicinity of the A-GDT. This condition was discovered by a health physics technician performing routine rounds in the primary auxiliary building. Upon discovery the area was posted as a high radiation area, health physics supervision was notified, and an ACTION Report was initiated to investigate the event. Based on a preliminary review, the health physics staff concluded that an unposted high radiation area existed in the GDT room for approximately one-half hour, and that no unplanned radiation exposures resulted from the event.

Health physics staff members stated that the gas transfer from the VCT to GDT was a routine event that occurred at the beginning of each refuel outage; the VCT to GDT gas transfer typically increased dose rates in the GDT room, but did not create a high radiation area; and the evolution was on the outage schedule. The operations manager and the health physics staff stated that this event occurred due to the presence of increased noble gas activity in the primary system (associated with known damage to several fuel pins), and due to a lack of communication between the operations and health physics organizations. The inspector noted that the licensee was aware that noble gas concentrations in the primary system were high due to leakage from several fuel pins, and had the opportunity to anticipate this event. The operations supervisor acknowledged the inspectors observation and stated that the ACTION Report was assigned to the operations group for resolution and that proposed corrective actions were to develop a procedural requirement for the operations group to communicate/coordinate future degassing of the VCT to the GDT with the health physics organization, and to evaluate other operational evolutions that had the potential for changing radiological conditions.

This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy **(NCV 50-244/97-11-05).**

c. Conclusions

Based on this review, the inspector made the following conclusions:

- Although corrective actions were appropriate, the licensee had the opportunity to anticipate that venting of the volume control tank to the gas decay tank would result in elevated radiation areas, potentially creating a high radiation area.

- The threshold for entering radiological control deficiencies into the ACTION

reporting system had decreased, and the system was effectively used to identify and resolve radiological control deficiencies.

R8 Miscellaneous RP&C Issues

R8.1 (Update) VIO 97-01-02: Contamination Boundary Controls

Radiological boundaries inside and outside of the plant were well defined and clearly delineated, and doors to locked high radiation areas were well posted and securely locked. Concerted efforts had been made to improve human performance and controls for work in and around radiological boundaries. This included development of revised administrative guidelines for work in and around radiological boundaries (project boundary); conduct of training on revised boundary controls; issuance of several management communications; and initiation of a work stand-down for several hours on two shifts during a refuel outage to emphasize the importance of complying with radiological boundary controls. In spite of these efforts, human performance deficiencies related to radiological boundaries continued to be identified at a relatively high rate. For example, project boundary guidelines did not allow individuals to reach across contaminated boundaries without health physics oversight. On November 4, 1997, the inspector observed an individual reaching into a posted contaminated area within a hydrogen analyzer cabinet on the lower elevation of the intermediate building without health physics oversight; on November 4, 1997, the inspector observed an individual exit the containment pressurizer cubicle without removing an outer layer of shoe covers at the step-off-pad (SOP); and on November 6, 1997, an ACTION Report was written after a quality assurance inspector observed a health physics technician reach across a contaminated area boundary without gloves (bare hand) to obtain a loose contamination smear. Although in each case, contamination levels were low and there were no significant consequences, the continued identification of deficiencies indicated that management had not been fully successful in improving human performance with regard to work in and around radiological boundaries.

R8.2 (Closed) URI 97-09-02: Potential Deficiency in Audit of Process Control Program

A surveillance of the radwaste/material program was performed on October 8-9, 1997. This surveillance included a review of the process control program and the use of software to evaluate the isotopic content of radioactive material/waste packages. The inspector concluded that performance of this surveillance satisfied the requirement documented in Quality Assurance Program for Station Operation, Rev. 23, dated December 13, 1996, for the process control program and implementing procedures to be audited on a 24-month cycle. This item is closed **(URI 50-244/97-09-02).**

R8.3 (Update) IFI 95-15-01: Inleakage Into the Residual Heat Removal (RHR) System Pump Room

The inspector reviewed efforts taken to evaluate and minimize fuel pool leakage into the RHR pump room. Information was gathered by a review of efforts to seal and

monitor the fuel pool transfer slot, efforts to evaluate fuel pool water balance, and efforts to monitor ground water tritium activity.

During the week of August 11, 1997, an epoxy seal was applied to the bottom of the fuel transfer slot where anchor bolts penetrated the liner. The inspector noted that at that time, a system that collected water from beneath the fuel pool and transfer slot drained into a sump located in the chemical and volume control system (CVCS) tank room at a rate of approximately 2 drops per second. On November 4, 1997, with the fuel transfer slot flooded, the inspector observed that the leakage rate from the same drain hose had increased to almost a steady stream. The inspector noted that licensee staff had not taken steps to monitor this leakage rate and had not sampled this water to compare the activity or boron concentration to the water in the fuel pool transfer slot.

The inspector also inquired if fuel pool water balance (e.g., fuel pool water makeup - water evaporation = leakage) was being pursued as a means to evaluate potential fuel pool leakage. The inspector was informed that due to potential error associated with these measurements, fuel pool water balance was not being pursued as a means to evaluate fuel pool leakage.

Ground water samples were being taken to evaluate tritium activity in ground water. No definite trends had been established to related current ground water tritium concentrations to fuel pool leakage or the formerly degraded steam generator blowdown piping.

The plant manager stated that the licensee was currently performing an evaluation to determine what future actions needed to be taken to evaluate and minimize fuel pool leakage.

Based on this information the inspector concluded that actions had been taken to seal potential water leakage pathways in the fuel transfer slot; that evaluation of fuel pool water balance was not being pursued as a method to evaluate fuel pool leakage; that ground water tritium activity was being monitored on a periodic basis; and additional plans to evaluate and minimize fuel pool leakage were being considered.

R8.4 Updated Final Safety Analysis Report (UFSAR) Review

The inspector reviewed Section 12.1, "Ensuring that Occupational Radiation Exposures are As Low As is Reasonably Achievable," and Section 12.5, "Radiation Protection Program Administration." No UFSAR discrepancies were identified during this review.

V. Management Meetings

X1 Exit Meeting Summary

The results of the radiological controls inspection were presented to the licensee on October 31, 1997, and the ISI/NDE inspection results were presented on November 7, 1997. After the entire inspection period was concluded, the resident inspectors presented the overall inspection results to members of licensee management on November 26, 1997. The licensee acknowledged the findings presented.

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

L2 Review of UFSAR Commitments

While performing the inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspector verified that the UFSAR wording was consistent with the observed plant practices, procedure and/or parameters.

ATTACHMENT I

PARTIAL LIST OF PERSONS CONTACTED

T. Alexander	Nuclear Assurance Manager
B. Flynn	Primary Systems Engineering Manager
C. Forkell	Electrical Systems Engineering Manager
G. Graus	I&C/Electrical Maintenance Manager
A. Harhay	Chemistry & Radiological Protection Manager
A. Herman	Health Physicist
J. Hotchkiss	Mechanical Maintenance Manager
G. Jones	Primary Chemist
G. Joss	Results and Test Supervisor
M. Lilly	Quality Assurance Manager
R. Marchionda	Production Superintendent
F. Mis	Principle Radiation Physicist
R. Ploof	Secondary Systems Engineering Manager
J. Smith	Maintenance Superintendent
J. Widay	Plant Manager
T. White	Operations Manager
G. Wrobel	Nuclear Safety & Licensing Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 60705:	Preparation for Refueling
IP 60710:	Refueling Activities
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 73753:	Inservice Inspection
IP 83750:	Occupational Radiation Exposure
IP 92901:	Follow-up - Operations
IP 92903:	Follow-up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

IFI 97-11-01	Foreign Material Exclusion Controls
IFI 97-11-02	Quality and Content of Radiological Briefings Provided to Plant Personnel
IFI 97-11-03	Use of Isotopic Scaling Factors for Internal Dose Assessment

IFI 97-11-04	Actions to Investigate and Document Refuel Cavity/Fuel Transfer Canal Leakage
NCV 97-11-05	Failure to Post a High Radiation Boundary

Closed

NCV 97-11-05	Failure to Post a High Radiation Boundary
URI 96-06-03	Modifications to Auxiliary Feedwater Valves MOV-4007 and MOV-4008
VIO 96-01-01	Both PORVs Inoperable During Stroke Time Adjustment
VIO 96-05-01	Failure to Establish Containment Refueling Integrity
URI 95-08-01	UFSAR Pertaining to ISI/NDE Program Not Consistent With ASME Code Section V
URI 95-08-01	Evaluate the Root Cause Analysis and Corrective Actions for MOV-4007 Failure

Discussed

IFI 97-10-01	Weak Configuration Controls
VIO 97-01-02	Contamination Boundary Controls
IFI 95-15-01	Inleakage Into the Residual Heat Removal Pump Room

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
BWI	Babcock and Wilcox International
CCW	Component Cooling Water
CVCS	Charging and Volume Control System
CW	Circulating Water
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
cpm	counts per minute
d/p	differential pressure
dpm	disintegrations per minute
ECCS	Emergency Core Cooling System
ECT	Eddy Current Test
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
FME	Foreign Material Exclusion
HPSI	High Pressure Safety Injection
IFI	Inspector Follow-up Item
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Test
ITS	Improved Technical Specification
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LTOP	Low Temperature Overpressure Protection
MDAFW	Motor-Driven Auxiliary Feedwater
MOV	Motor-Operated Valve
MRPC	Motorized Rotating Probe Coil
MS	Main Steam
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Test
NDE	Nondestructive Examination
NQI	Non-Qualified Indication
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSARB	Nuclear Safety Audit and Review Board
PORC	Plant Operations Review Committee
PORV	Power-Operated Relief Valve
ppm	parts per million
PT	Periodic Test
psig	pounds per square inch gage
PWR	Pressurized Water Reactor
QA	Quality Assurance

QC	Quality Control
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RG&E	Rochester Gas and Electric Corporation
RHR	Residual Heat Removal
RF	Refueling Procedure
RP	Radiation Protection
RP&C	Radiological Protection and Chemistry
RSSP	Refueling Shutdown Surveillance Procedure
RTD	Resistance Temperature Detector
RWP	Radiation Work Permit
RSG	Replacement Steam Generator
SAFW	Standby Auxiliary Feedwater
SG	Steam Generator
SEV	Safety Evaluation
SI	Safety Injection
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SW	Service Water
Tave	Average RCS Temperature
TDAFW	Turbine-Driven Auxiliary Feedwater
TPCN	Temporary Procedure Change Notice
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Test
VIO	Violation