U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No.	50-244/97-09
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:	August 4, 1997 through September 7, 1997
Inspectors:	P. D. Drysdale, Senior Resident Inspector C. C. Osterholtz, Resident Inspector 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 50-244/97-09

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 radiation specialist.

Operations

The Ginna plant remained at full power throughout the inspection period. All operator performance observed throughout the inspection period was good. Operators demonstrated effective communications and performed actions in accordance with procedural requirements.

The procedure used to verify containment spray system status was adequate. Auxiliary operator performance was considered good, and the system valve positions were verified to be consistent with the plant piping and instrumentation drawings (P&IDs).

The program to track plant jumpers and lifted leads was lacking in that one lifted lead was not adequately tracked in accordance with program requirements. The inspector considered that a single document to provide control room operators with the status of plant jumpers and lifted leads could enhance the tracking process. The licensee acknowledged the issue and indicated that program improvements would be investigated.

Several instances of poor verbal cuing techniques had been employed during the administration of observed job performance measures (JPMs), and poor cuing contributed to the failure of an operator to successfully perform a JPM task. Pending further inspection in this area, a new inspector follow-up item was opened **(IFI 50-244/97-09-01)**.

Licensee Event Report (LER) 97-003 was submitted in response to the licensee's discovery that all four high steam flow inputs to the main steam isolation circuitry were inoperable. The LER adequately described the event and appropriately addressed the root causes and corrective actions.

Maintenance

Observed maintenance activities 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 during the maintenance work.

Surveillance procedures were current and properly followed, and all surveillance work was properly authorized. 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 (UFSAR).

Executive Summary (cont'd)

The maintenance work to seal the bottom of the fuel transfer canal was well planned and executed. Radiological protection (RP) controls associated with the work were effective, and no spread of contamination or unexpected personnel exposure resulted from the work.

The licensee's determination that debris found in check valve CV-1810A and CV-1810B was flushed from the refueling canal drains appeared likely, given the low radioactive level of the debris, and the fact that the refueling canal drains were not closed during refueling transfer canal clean up after the previous refueling outage. The licensee's intentions to analyze other systems for foreign material exclusion (FME) deficiencies and to increase emphasis on FME concerns in training were appropriate.

Instrumentation and control (I&C) technicians successfully identified the cause of the failure of power range channel N-43 and effectively repaired the instrument. The replacement parts did not completely reflect those that had been previously installed. The licensee appropriately generated an ACTION Report to investigate the issue.

Engineering

Additional thermal performance tests of the EDG coolers provided adequate justification that cooler performance did not degrade to an inoperable condition over the current operating cycle, and that visual inspections of the heat exchangers during the 1997 refueling outage would be adequate to meet the requirements of the licensee's GL 89-13 program. However, pending additional confirmation of the actual fouling factors with the cooler manufacturers, and a NRC review of the licensee's analyses of other safety-related service water heat exchanger performance, item **URI 50-244/96-06-04** remains open.

The licensee requested that Westinghouse perform a sensitivity analysis for the Ginna large break loss of coolant accident (LOCA), assuming that emergency core cooling system (ECCS) leakage was 2.75 gallons per hour (gph). The results of the analysis indicated that leakage from the containment spray (CS) system test line should be limited to 2.75 gph following a large break LOCA, and that a 2.75 gph leak would not produce offsite or control room exposures in excess of 10 CFR 100 limits. The licensee revised the basis for LCO 3.6.1, "Containment," and incorporated the higher ECCS leakage allowance into the Improved Technical Specification Bases and plant procedures. All of the follow-up actions related to this item were satisfactory. Item **URI 50-244/96-07-01** was closed. Revision 2 to LER 96-009 accurately reflected the outcome of all related issues.

Both auxiliary feedwater flow control valves MOV-4007 and MOV-4008 were modified in September 1996 to restore the valve plugs to their original design, and to make the valves more reliable in achieving their necessary post-accident throttling requirements. Repeated tests on both valves resulted in final throttle flows in the range of 200 - 230 gpm as required. The licensee's actions satisfactorily resolved this issue. Item **URI 50-244/96-07-02** was closed.

Executive Summary (cont'd)

Plant Support

Procedures for handling, packaging, classifying, and shipping radioactive materials/wastes were well maintained and of good quality, and copies of appropriate state and federal regulations and licenses for facilities were current and well maintained. Elements of the process control program appeared effectively managed. Efforts to reduce the generation of radioactive waste (radwaste) volume were effective, and radioactive materials were stored in a neat and orderly fashion.

Radioactive material shipments had been accurately classified and shipped in accordance with Department of Transportation (DOT) and NRC regulations.

Radiological boundaries were well maintained and clearly delineated, and conditions of housekeeping were generally good.

Appropriate training on regulations and procedures pertaining to radioactive waste handling, processing, packaging, and shipping had been provided to the radwaste staff; staff members were qualified to perform tasks assigned within the radioactive material/waste management and transportation program; radwaste personnel demonstrated very good knowledge; and training programs were being effectively managed. Clear and informative training was developed to improve worker performance with regard to working in and around radiological boundaries.

Overall quality assurance, quality control, and self assessment program oversight of the radioactive material/waste transportation program were effective. However, the review of the process control program documented in the 1996 and 1997 audits was noticeably shallow, and did not include detail reviews in this area. This item is being treated as an unresolved item pending further review of audits and self assessments of the process control program. **(URI 50-244/97-09-02)**

Inspector follow-up item IFI 50-244/97-03-01 had been opened because a material inventory had not been maintained for the upper radwaste storage facility. In addition to existing radiological controls, a current inventory for the building was implemented and maintained. The inventory was reflective of the building contents. Item **IFI 50-244/97-03-01** was closed.

The licensee's fire protection systems and capabilities were adequately supported by licensing basis documents, although some minor inconsistencies were noted. Operator actions for responding to a control room fire were adequately implemented by plant procedures and simulator training. The fire brigade was provided with appropriate procedures, protective equipment, and training to deal with a control room fire.

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ATTACHMENT

Attachment 1 - Partial List of Persons Contacted
- Inspection Procedures Used
- Items Opened, Closed, and Discussed
- List of Acronyms Used

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: 1) tours of the accessible areas of the facility; 2) 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; 3) verification of engineered safety feature (ESF) system operability; 4) verification of proper control room and operator shift staffing; and 5) verification that logs and records accurately identified equipment status or deficiencies.

O1.2 Summary of Plant Status

The Ginna plant remained at full power throughout the inspection period. All operator performance observed throughout the inspection period was good. Operators demonstrated effective communications and performed actions in accordance with procedural requirements.

Offsite power was temporarily transferred to a 100/0 lineup on circuit 767 on August 13 and 15, 1997, when circuit 751 was considered vulnerable to thunderstorms. When returning to the 50/50 alternate offsite power lineup on August 13, the bus 12B alternate feed breaker from circuit 751 did not close on the first attempt. Electrical technicians inspected the system and found no discrepancies, and the breaker successfully closed on the second attempt. The licensee generated an ACTION Report (97-1224) to document the anomaly.

Technical support center ventilation was secured for approximately two weeks beginning August 15, 1997, due to a failed air supply motor. No ITS limiting condition for operation (LCO) applied to this condition; however, the inspector verified that the licensee had developed appropriate procedures to evacuate the TSC if it becomes uninhabitable following its activation. Additionally, the control room emergency air treatment system (CREATS) actuation instrumentation was out of service for approximately one week beginning August 31, 1997, due to a power supply failure in R-38, the control room iodine monitor. The licensee placed the control room heating, ventilation, and air conditioning (HVAC) system into its recirculation mode (Mode F) within one hour, as required by LCO 3.3.6. The control

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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.

room HVAC system has not met its maintenance rule performance criteria since the first quarter of 1996.

On August 24, 1997, control room operators entered procedure AP-RCC.1, "Continuous Control Rod Withdrawal/Insertion," in response to control rods automatically inserting due to power range channel N-42 failing high. N-42 was taken out of service and rods were returned to automatic control. Instrumentation and control (I&C) technicians subsequently discovered a failed test signal potentiometer on N-42. The potentiometer was replaced and N-42 was returned to service the same day. On September 3, 1997, control room operators noted that power range channel N-43 axial flux was slowly increasing, and subsequently declared it inoperable. The operators promptly notified I&C maintenance, and subsequent troubleshooting revealed high resistance in wiring connecting two current gain potentiometers. N-43 was repaired and returned to service the same day (see section M2.3).

At the end of the report period, the inspectors noted that packing leakage on the C-charging pump had increased to its administrative limit of 0.5 gallons per minute (gpm). The leakage was reduced below the administrative limit by placing the pump in the manual mode of operation. The licensee indicated that the C-charging pump packing would be replaced soon after the next refueling outage (October - November, 1997) since this work was not planned before the outage scope was fixed. However, replacement parts are available onsite, and the licensee is prepared to rebuild the pump before then if the leakage becomes significant.

O2 Operational Status of Facilities and Equipment

O2.1 Containment Spray System Walkdown

a. Inspection Scope (71707)

The inspector observed an auxiliary operator (AO) perform a walkdown of the containment spray system.

b. Observations and Findings

The inspector observed the AO perform system procedure S-30.3, "Containment Spray System Valve and Breaker Position Verification," on August 22, 1997. This procedure was performed to complete a monthly verification that the containment spray system adequately reflected the designated normal system lineup. The AO correctly identified all associated valves and their status during the walkdown. The inspector verified that normal system status was consistent with that indicated in the piping and instrumentation drawings (P&IDs).

c. Conclusions

The inspector concluded that the procedure used to verify containment spray system status was adequate. Auxiliary operator performance was considered good, and the valve positions were consistent with those indicated on plant P&IDs.

O2.2 Control of Jumpers and Lifted Leads

a. Inspection Scope (71707)

The inspector reviewed the licensee's program for maintaining control of jumpers and lifted leads installed throughout the plant.

b. Observations and Findings

On September 2, 1997, the inspector reviewed procedure A-25.7, "Bypass of I&C/Electrical Function and Jumper Control on Out of Service Equipment," and the associated shift supervisor's A-25.7 tracking book. The inspector noted multiple occurrences in the A-25.7 tracking book where paperwork had apparently been cleared but was not indicated as such on the tracking book's cover page log sheet. Additionally, one active jumper listed in the log did not have an estimated removal date. The inspector also noted that there was no one document available to control room operators indicating the status of all the jumpers and lifted leads in the plant, as some jumpers/lifted leads were installed as temporary modifications, others in accordance with an existing maintenance or surveillance procedure, and some in accordance with the A-25.7 procedure. The shift supervisor's status book for temporary modifications contained some information on jumpers and lifted leads, but that information was not comprehensive. Also, the control room maintained an active list of main control board annunciators that were affected by a jumper or lifted lead. However, the status of all jumpers and lifted leads could not be determined from the information maintained in the control room. The licensee generated an ACTION Report (97-1382) to address this issue, and subsequently discovered that a lifted lead for the J-31 annunciator for the B-vital battery monitor had not been tracked as required by A-25.7. Consequently, the licensee generated another ACTION Report (97-1384) to address this issue.

c. Conclusions

The inspector concluded that the program to track plant jumpers and lifted leads was lacking in that one lifted lead was not adequately tracked in accordance with program requirements. The inspector considered that a single document to provide control room operators with the status of plant jumpers and lifted leads could enhance the tracking process. The licensee acknowledged this issue and indicated that program improvements would be investigated.

O5 Operator Training and Qualification

O5.1 In-Plant Job Performance Measure (JPM) Administration

a. Inspection Scope (71707)

The inspector observed the administration of in-plant JPMs performed as part of the licensed operator requalification program.

b. Observations and Findings

On September 3, 1997, the inspector observed three in-plant JPMs administered to a senior reactor operator (SRO) candidate by the licensee's training department as part of his annual requalification examination. The inspector noted some confusion between the examiner and the candidate on one of the JPMs that the candidate subsequently failed. The examiner had provided inappropriate verbal cues in response to the actions the candidate had taken, leading the candidate to believe he had performed actions correctly, when in fact he had not.

In addition, the inspector noted several minor cuing deficiencies. For example, several instances occurred where the candidate needed to obtain information from local plant gages, and the examiner did not provide the actual values he would have seen. The licensee considered the JPM to be invalid for the candidate since it had been administered improperly, but indicated that no administrative problems occurred when the JPM was administered on previous candidates. The inspector observed the candidate perform an additional in-plant JPM that was administered by a different examiner, and the candidate passed the JPM with no discrepancies noted. The licensee generated an ACTION Report (97-1385) to address the invalid JPM.

c. Conclusions

The inspector concluded that several instances of poor verbal cuing techniques had been employed during the administration of the observed JPMs, and that poor cuing contributed to the failure of an operator to successfully perform a JPM task. The inspectors considered that similar potential weaknesses could exist in the training department's administration of JPMs. Therefore, pending further inspection in this area, an inspector follow-up item was opened (IFI 50-244/97-09-01).

O8 Miscellaneous Operational Issues

O8.1 (Closed) Licensee Event Report (LER) 97-003: Bistable Instrument Setpoint (Plus Instrument Uncertainty) Could Exceed Allowable Value, Causes a Condition Prohibited by Plant Technical Specifications

LER 97-003 was submitted on August 27, 1997, in response to the licensee's discovery that all four high steam flow inputs to the main steam isolation circuitry were inoperable because the existing channel bistable setpoint would not ensure

that the ITS "trip allowable value" would be met after incorporation of the setpoint's uncertainty and drift (see IR 50-244/97-06). The inspectors considered that the LER adequately described this event and appropriately addressed the root causes and corrective actions. This LER is closed (LER 97-003).

II. Maintenance

M1 Conduct of Maintenance

M1.1 Observations of Maintenance Activities

a. Inspection Scope (62707)

The inspectors observed portions of plant maintenance activities to verify conformance with the maintenance rule (10 CFR 50.65), that the correct parts and tools were utilized, that the applicable industry codes and ITS requirements were satisfied, that adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and that the licensee properly verified equipment operability upon completion of post maintenance testing.

b. Observations and Findings

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

- CPI-MON-R17, "Calibration of Radiation Monitoring System Component Cooling Water Monitor R-17;" observed on August 13, 1997.
- Fuel transfer canal sealing; observed on August 14, 1997 (see section M2.1)
- PM-500001, "MOV Grease Check and Stem Lubrication," for MOV-4007, A-MDAFW discharge valve; observed on August 19, 1997.
- CMP-10-03-EAF02A, "American Standard, Model SSCF, Heat Exchanger Maintenance for EAF02A," for the A-AFW lube oil cooler; observed on August 19, 1997. This maintenance was performed in conjunction with MOV-4007 maintenance. The lube oil cooler was removed cleaned, inspected, and replaced. No discrepancies were noted.
- WO #19603605, "Perform Major Inspection on CV-1810A," for the A-containment sump suction check valve to the A-reactor coolant drain tank (A-RCDT) pump; observed on August 19, 1997 (see section M2.2).
- CPI-Axial-N43, "Calibration of Power Range Instrument N-43;" observed on September 3, 1997, following N-43 troubleshooting and repair (see section M2.3).

- CPI-MON-R15, "Calibration of Radiation Monitoring System Steam Generator Blowdown Monitor R-19;" observed on September 4-5, 1997. This annual calibration included installation of a jumper to prevent automatic steam generator blowdown isolation while R-19 was being calibrated. The installation and removal of the jumper was successfully performed per the calibration procedure, but was not required to be included in the shift supervisors tracking log for jumpers and lifted leads (see section O2.2).

- WO #19702478, "Spent Fuel Pool Charcoal Filter Reconfiguration;" observed on September 2-3, 1997.

- WO # 19702296, A-penetration cooling fan disassembly, repair, and reassembly; observed on September 2-3, 1997.

c. Conclusions

The inspectors concluded that the observed maintenance activities were performed in accordance with procedure 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 Observations of Surveillance Activities

a. Inspection Scope (61726)

The inspectors observed selected surveillance tests to verify that approved procedures were in use and procedure details were adequate, that test instrumentation was properly calibrated and used, that ITS surveillance requirements were satisfied, that testing was performed by qualified and knowledgeable personnel, and that test results satisfied acceptance criteria or were properly dispositioned.

b. Observations and Findings

The inspectors observed portions of the following surveillance activities:

- PT-12.1, "Diesel Generator A" with SW test equipment installed; observed on August 7, 1997 (see section E2.1)

- PT-12.2, "Diesel Generator B" with SW test equipment installed; observed on August 12, 1997 (see section E2.1)

- PT-2.8Q, "Component Cooling Water Pump Quarterly Test;" observed on August 13, 1997, for the A- and B-CCW pumps.

- PT-1, "Rod Control System;" observed on September 2, 1997. This surveillance verified the operability of each of the full length control rod assemblies, the rod drive mechanism, and the multiplexing rod position

indication (MRPI) system by moving each rod control cluster assembly a sufficient number of steps to cause a change of position as indicated by the MRPI system. No discrepancies were noted during the test.

- PT-33A, "A Spent Fuel Pool Pump;" observed on September 3, 1997.
- PT-32B, "Reactor Trip Breaker Testing - B Train;" observed on September 5, 1997.

c. Conclusions

The inspectors confirmed that the procedures used were current and properly followed, and that the shift supervisor properly authorized all surveillance work to proceed. The licensee certified 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.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Fuel Transfer Canal Sealing

a. Inspection Scope (62707)

The inspectors observed maintenance activities related to sealing the bottom surfaces of the fuel transfer canal as an effort to prevent spent fuel pool leakage from the transfer canal into the RHR pump room (see IRs 50-244/95-17, 50-244/97-05, and 50-244/97-06).

b. Observations and Findings

On July 7, 1997, the licensee replaced the spent fuel pool (SFP) weir gate bladder to reduce the amount of SFP water leaking into the transfer canal. SFP water had been collecting in the canal and was apparently leaking out through hardware connections in the bottom of the canal and into the RHR pump room. A contractor had made a previous attempt in 1996 to seal the transfer canal, but all hardware connections were not covered and water leakage into the RHR pump room was not completely stopped.

During the week of August 11, 1997, the licensee dried out and cleaned the bottom of the canal in preparation for the contractor to reapply the sealant to all hardware connections. During the same week, the licensee made preparations to receive the contractor's equipment onsite, and to plan the radiological controls associated with the work. Past refueling operations have caused the fuel transfer canal to be controlled as a high radiation and high contamination area. Prejob surveys indicated radiation levels up to approximately 160 mR/hr at waist level, and 1400 mR/hr at ankle level. The radiological protection (RP) organization made preparations that included provisions for preventing a high airborne contamination problem while the

work was in progress. These included a nylon drape over the transfer canal slot, a portable high capacity high efficiency particulate air (HEPA) filter and ventilation line, and full protective clothing for workers.

On August 13, 1997, the licensee conducted a detailed prejob briefing with maintenance, engineering, RP, and contractor personnel to review all work controls and to discuss all concerns associated with the work. On August 14, 1997, the licensee restricted general access to the auxiliary building before and during progress of the work. Also, the auxiliary building ventilation system was shut down to protect the spent fuel pool charcoal beds from any volatile chemicals from the sealant (isocyanate and resin mixture). Throughout that day, the sealant was applied with a spray gun mixing apparatus. Minor equipment problems caused a delay in completing the work within the expected time; however, the problems were corrected without significant difficulty, and all of the sealing work was completed that day. Surveys conducted after the sealing work was completed indicated that no spread of contamination occurred outside of the transfer canal, or on any of the contractor's equipment leaving the site. The total personnel exposure for the job was less than the licensee had anticipated.

c. Conclusions

The inspectors concluded that the maintenance work to seal the bottom of the fuel transfer canal was well planned and executed. RP controls associated with the work were effective, and no spread of contamination or unexpected personnel exposure resulted from the work.

M2.2 A-Containment Sump Check Valve (CV) to the A-Reactor Coolant Drain Tank (A-RCDT) Pump, CV-1810A, Preventive Maintenance Inspection

a. Inspection Scope (62707)

The inspector observed licensee maintenance personnel perform a preventive maintenance (PM) inspection on check valve CV-1810A.

b. Observations and Findings

On August 19, 1997, the inspectors observed maintenance personnel dismantle CV-1810A for inspection, cleaning, and lubrication. The RCDT collects waste leakage through CV-1810A from the RCS loop drains, No. 2 seal leakoff from the reactor coolant pumps (RCPs), excess letdown, reactor vessel flange leakoffs, safety injection (SI) system accumulator drains, pressurizer relief tank (PRT) drains, refueling canal drains, and the spray valves. All drains to the RCDT are closed drains, with the exception of the fuel transfer canal drains. The RCDT and the RCDT pumps are nonsafety-related. This maintenance activity is scheduled on a 54 month frequency.

After CV-1810A was disassembled, maintenance personnel noted foreign material laying at the bottom of the valve housing. Several small washers, a small screw,

several small nuts, some broken pieces of plastic and a plastic tie wrap, and several pop rivets were included in the debris. The licensee generated an ACTION Report (97-1254) to address this issue. The foreign materials were removed and the valve was inspected and reassembled. The licensee subsequently inspected CV-1810B, the A-containment sump suction check valve to the B-RCDT pump, on August 21, 1997. The licensee discovered similar debris in the valve housing, but it was a smaller quantity. The foreign material was removed, and the valve was inspected and reassembled.

The licensee concluded that small debris may have been flushed from the refueling transfer canal drains after the 1996 refueling outage because the canal drains were not closed during the refueling cavity clean up. Also, the low radioactive levels detected on the foreign material strongly suggested that the debris did not come from the RCS. The licensee indicated that they intended to analyze other systems for the potential of foreign material exclusion (FME) deficiencies; however, an ACTION Report had not been initiated prior to the end of the report period. The licensee also indicated that increased emphasis will be placed on preventive FME work practices during maintenance worker training.

c. Conclusions

The inspector concluded that the licensee's determination that debris found in CV-1810A and CV-1810B was flushed from the refueling canal drains appeared likely, given the low radioactive level of the debris, and the fact that the refueling canal drains were not closed during refueling transfer canal clean up after the previous refueling outage. The licensee's intentions to analyze other systems for FME deficiencies and to increase emphasis on FME concerns in training were appropriate.

M2.3 Power Range Channel N-43 Troubleshooting and Repair

a. Inspection Scope (62707)

The inspector observed I&C personnel perform troubleshooting and repairs on nuclear instrument power range channel N-43.

b. Observations and Findings

At approximately 4:00 a.m. on September 3, 1997, operations personnel noted that the N-43 axial flux was slowly increasing with all other power range instruments indicating normal. Since the trend continued, the licensee declared N-43 inoperable at 8:58 a.m. and removed it from service. The licensee also generated an ACTION Report (97-1379) to document the issue. While troubleshooting, I&C technicians discovered high resistance readings in wiring connecting two current gain adjustment potentiometers. The technicians concluded that the solder connections between the wires and the potentiometers had deteriorated and decided to replace the potentiometer's wires.

The technicians noted that the "fine" current gain potentiometer procured from stock did not have a screw adjustment slot to adjust the potentiometer, whereas the previously installed one did. The technicians suggested that the previously installed potentiometer's screw slot may have been added in the field after procurement, as it appeared off-center and discolored. The technicians also noted that two "course" current gain potentiometers procured from stock had different configurations, one with a screw slot and one without. The previously installed course adjustment potentiometer had a screw slot that appeared to be factory machined. The technicians replaced the course adjustment potentiometer with the replacement part that had a screw slot, as required by design specifications. They also replaced the "fine" current gain potentiometer, realizing that adjustments made to that potentiometer would have to be performed by hand. The licensee generated an ACTION Report (97-1372) to address the question of potentiometer screw slots, and to determine if an improper course adjustment potentiometer had been procured and entered into the plant's spare parts. The screw slots do not have any effect on the electronic function of the potentiometers; however, both the fine and course adjustment potentiometers were procured as safety-related components. The N-43 instrument rack was reinstalled, calibrated, and returned to service at 8:35 p.m. on September 3, 1997.

c. Conclusions

The inspectors concluded that I&C technicians successfully identified the cause of the failure of power range channel N-43 and effectively repaired the instrument. However, the inspectors were concerned that the replacement parts were procured as safety-related components, but they did not completely reflect those that had been previously installed. Generating an ACTION Report to investigate the issue was appropriate.

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Updated) Unresolved Item (URI) 50-244/96-06-04: Thermal Performance Test Program for Service Water Heat Exchangers

URI 50-244/96-06-04 was opened in July 1996 to review engineering's planned and ongoing actions to complete the initial reviews of service water thermal performance tests that the licensee committed to perform in response to NRC Generic Letter (GL) 89-13. The initial service water heat exchanger thermal performance test program had not been completed at that time. For the 18 sets of service water heat exchanger thermal performance tests performed prior to July 1996, only the first (baseline) test of the emergency diesel generator (EDG) coolers had been fully evaluated with an engineering report written. In addition, the principal GL 89-13 program document had not been revised since September 1993. For some heat exchangers, the "outside" fouling factors were missing, and the maximum number of plugged tubes had not been determined.

During reviews of service water heat exchanger performance results, the licensee later considered that the initial baseline tests (April 1996) of both EDG coolers were adequate only for operability justifications. These test results were not considered adequate to justify only performing periodic visual inspections to confirm that the heat transfer capability of the coolers would satisfy GL 89-13 program requirements. The initial reports used test data that was significantly scattered, and did not contain and uncertainty factors for the temperature and flow instruments, or for the derived fouling factors. In addition, the change in cooler performance over a plant operating cycle was not determined.

All EDG coolers were completely retubed during the 1996 refueling outage, and the licensee considered that the change in thermal performance over an operating cycle should be obtained for all coolers. The additional tests were performed during the normal monthly EDG surveillance tests on August 7 and 12, 1997. Resistance temperature detectors (RTDs) were used to measure inlet and outlet service water and component cooling water in the diesel water jack and lube oil coolers. Ultrasonic (Controlotron) detectors were used to measure coolant flows. The inspectors observed both of these performance tests and reviewed the results detailed in the post-test analysis reports (attachments to tests PT-60.4 and PT-60.5).

For analyzing cooler performance, the licensee utilized improved computer software (TKSolver) that better incorporated actual heat exchanger design features into the analyses, and was capable of determining the maximum number of tubes that could be plugged. The analyses resulted in a total uncertainty of approximately 20% for the lube oil cooler fouling factors. The inspectors noted that PT-60.4 also contained some service water flow anomalies when the Controlotron flow detector provided fluctuating and unreliable data after the A-EDG was started. However, the licensee concluded that service water flow during that test could be accurately determined from flow data immediately after the engine was stopped, which was stable and consistent with service water flow in the B-EDG.

The test results and the computer analyses indicated that the A-EDG lube oil cooler maximum fouling factor (0.00767 hr-ft^2-°F/BTU) was approximately five times the maximum value stated by the cooler manufacturer, and the maximum fouling factor for the B-EDG lube oil cooler (0.00619 hr-ft^2-°F/BTU) was four times higher. The jacket water coolers for both the A- and B-EDGs had negative fouling factors, (-0.00108 hr-ft^2-°F/BTU and -0.00107 hr-ft^2-°F/BTU, respectively) which indicated to the licensee that both coolers were oversized by design. The licensee considered that the software used for the analyses may have made different assumptions regarding the heat exchanger design. However, the test results indicated that the cooler bypass flows for the lube oil coolers at full engine load were approximately 55% for the A-EDG, and approximately 46% for the B-EDG cooler. Also, the overall heat transfer rates calculated for each of the coolers was consistent with the design values provided from the manufacturer. These parameters provided confirmation that both coolers had a sufficient thermal margin to conclude that their performance was adequate.

The licensee intended to contact the cooler manufacturer in the future regarding the differences between the design and the derived fouling factors. However, the licensee concluded that the test results and analyses confirmed that the EDG coolers do not foul to the point of being inoperable over one plant operating cycle, and further determined that routine thermal testing of these coolers is not warranted. The inspectors agreed with the licensee's conclusions, and that visual inspections of the heat exchangers during the 1997 refueling outage appeared to be adequate to meet the requirements of the licensee's GL 89-13 program. However, pending additional confirmation of the actual fouling factors with the cooler manufacturers, and NRC review of all the licensee's analyses of the other safety-related service water heat exchanger thermal performance, this item remains open (URI 50-244/96-06-04).

E8.2 (Closed) URI 50-244/96-07-01: Containment Spray (CS) Test Line Leakage

URI 50-244/96-07-01 was opened in August 1996 because the plant was operated outside an analyzed "program limit" prior to isolating containment spray system leakage on July 22, 1996. The leak rate was small, but could not be isolated without declaring all emergency core cooling system (ECCS) trains inoperable. The plant was operated for approximately 20 hours while the leakage was evaluated. The following day, limiting condition for operation (LCO) 3.0.3 was entered pursuant to administrative procedure A-52.4, "Control of Limiting Conditions for Operation," and periodic test PT-39, "Leakage Evaluation of Primary Coolant Sources Outside Containment," which contained requirements to enter LCO 3.0.3. if ECCS leakage exceeded 2 gallons per hour (gph). These requirements were based upon Chapter 15 of the Ginna UFSAR (Section 15.6.4.3 and Table 15.6-21) which contained the 2 gph limit. The CS leakage measured during the event was 2.599 gph, and the total integrated ECCS leakage was 2.742 gph. The licensee entered LCO 3.0.3. after the leakage was measured and determined to be above the limit, and could not be isolated.

The licensee subsequently requested Westinghouse perform a sensitivity analysis for the Ginna large break loss of coolant accident (LOCA), assuming that ECCS leakage was 2.75 gph. The results of the analysis were outlined in a letter to RG&E dated November 1, 1996, which indicated that leakage from the CS test line should be limited to 2.75 gph following a large break LOCA, and that a 2.75 gph leak would not produce offsite or control room exposures in excess of 10 CFR 100 limits. Based on the results of the Westinghouse analysis, the licensee revised the basis for LCO 3.6.1, "Containment," on February 11, 1997, and incorporated the higher ECCS leakage allowance. The revised basis indicated that the containment spray, safety injection, and residual heat removal systems must be considered for containment operability since the systems act as an extension of containment during the recirculation phase of an accident. Under LCO 3.6.1, excessive ECCS leakage would require entry into a timed action statement for containment operability, and would not require immediate entry into LCO 3.0.3. Leakage beyond 2.75 gph would first require declaring the containment inoperable, allowing one hour to restore its, or be to in Mode 3 in three hours and Mode 5 in 36 hours.

The licensee also revised procedure A-52.4 to change its reference from LCO 3.0.3 to LCO 3.6.1 if integrated ECCS leakage exceeded 2.75 gph. Test procedure PT-39 was revised on February 12, 1997, which incorporated the 2.75 gph limit into the test acceptance criteria. The licensee had also initiated a change request to change to leakage limit in UFSAR Chapter 15. However, that change is still pending until the next overall UFSAR revision, which is currently scheduled for May 1998. The inspector considered all of the above actions to be satisfactory. Item URI 50-244/96-07-01 is closed .

E8.3 (Closed) LER 96-009, Revision 2: Leak Outside Containment, Due to Weld Defect, Results in Leak Rate Greater Than Program Limit

On August 11, 1997, the licensee submitted Revision 2 to LER 96-009. The revision was necessary because of a NRC identified error in Revision 1 to that LER (see IR 50-244/96-09) which stated that the subject leaking CS pipe joint was only susceptible to carrying radioactive containment sump fluids following a small break loss of coolant accident (LOCA) during "high-head" recirculation. However, following a large break LOCA, the licensee's emergency procedure ES-1.3, "Transition to Cold Leg Recirculation," directed operators to reconfigure the RHR system in a manner that the CS pipe joint would also be subjected to full RHR pump discharge pressure with radioactive containment sump fluids. Under large break LOCA conditions with significant fuel damage, the maximum allowable CS leakage outside containment would be significantly less than was stated in Revision 1 of the LER (30 gph) for a small break LOCA. Therefore, the maximum leakage allowable for this joint (or any ECCS piping outside containment) had to be revised down to 2.75 gph to comply with a more recent Westinghouse analysis for a large break LOCA.

The issues related to the maximum allowable ECCS leakage outside containment were resolved under URI 50-244/96-07-01 above. Revision 2 to LER 96-009 accurately reflected the outcome of all related issues. Based upon the satisfactory conclusion of that item, LER 96-009 is closed.

E8.4 (Closed) URI 50-244/96-07-02: Minimum Auxiliary Feedwater (AFW) Flow Requirements for Accident Conditions

On August 20, 1996, a faulted control signal caused the B-main feedwater (B-MFW) regulating valve to go full closed, and the reactor automatically tripped when the B-steam generator level reached the low trip set point. Immediately after the reactor tripped, the AFW throttle valves did not throttle as stated in safety evaluation SEV-1073 and ITS basis 3.7.5. to ≥200 gpm. Several other instances occurred during AFW discharge valve testing where accurate and repeatable throttle flows were not achieved. The unreliable automatic throttling of these valves had also been exacerbated because an apparent design deficiency (see NRC IR 50-244/96-06, sections O2.2, E.1.1. and M1.2).

The licensee ran several computerized accident scenarios assuming that no AFW flow was available for 10 minutes after a reactor trip from Mode 1. In all cases, the

steam generators (SGs) did not achieve a dryout condition prior to 10 minutes. The licensee considered that the previous procedural guidance for operator action in one minute to establish ≥ 200 gpm total AFW flow was sufficient for AFW operability. The licensee stated that these results and additional considerations for AFW accident requirements would be clarified in a future revision to safety evaluation SEV-1073, and in a revision to the ITS bases for AFW.

The licensee issued Revision 1 to SEV-1073 on September 6, 1996, which evaluated several different accident scenarios for which AFW flow is credited, including a loss of main feed water with no reactor trip, and a SG tube rupture. The SEV concluded that a 10 minute delay for AFW injection into the SGs is acceptable for either the motor-driven or turbine-driven AFW pumps. Revision 6 to the ITS bases was issued on February 11, 1997, and incorporated the results of the revised SEV into basis section 3.7.5. The revision stated that one basis for AFW system operability included two motor-driven AFW trains capable of supplying their respective SG with ≥200 gpm within 10 minutes and ≤230 gpm upon AFW actuation.

Both MOV-4007 and MOV-4008 were modified in September 1996 to restore the valve plugs to their original design, and to replace actuator gearing. The modifications made the valves more reliable in achieving the necessary throttling conditions. The valves were subsequently tested under dynamic differential pressure conditions, starting from the open position with maximum AFW pump flow through the valves, and with zero pressure in the steam generators downstream of the valves. Repeated tests on both valves resulted in final throttle flows in the range of 200 - 230 gpm. The inspector considered that all the above results satisfactorily resolved this issue. Item **URI 50-244/96-07-02** is closed .

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Solid Radioactive Waste (Radwaste) Program

a. Inspection Scope (86750)

A review was performed of the solid radwaste program. Information was gathered by a review of procedures, copies of applicable regulations, licenses for facilities where radioactive materials were shipped to, use of scaling factors for the classification of wastes, through discussions with cognizant personnel, and tours through the facility.

b. Observations and Findings

Current copies of management approved procedures for handling, packaging, classifying, and shipping radioactive wastes were maintained in the radwaste field office. Procedures were clear, detailed, and of good quality.

Copies of applicable federal and state regulations were maintained in the radwaste field office. A contract had been established with a vendor to provide periodic updates, and a selected review revealed that copies of regulations were current and well maintained.

Copies of facility and state licenses for facilities where radioactive materials and wastes were shipped to were maintained in binders in the radwaste field office. These binders were readily available and well organized and labeled. A selected review revealed that licenses were current or within a timely renewal status. Procedural guidance also required verification that radioactive material licenses for the consignee of shipments to be on-file. No discrepancies were identified.

Procedures outlining the Process Control Program (PCP) and methods for classifying wastes in accordance with 10 CFR 61 were available and provided clear and reasonable guidance. Attempts to obtain representative samples from various waste streams included periodic sampling of sluiced resins, selected sampling of cartridge filters, and smear sampling. An independent laboratory was used to evaluate the isotopic content of various waste streams such as dry active waste (DAW), and liquid waste processing (LWP) bead resin. Documentation revealed that procedurally required sample frequencies were being met and that data validations were performed. The validated data was used to develop scaling factors to relate the concentration or abundance of easily measurable nuclides such as gamma emitters to hard to detect nuclides such as alpha and low energy beta emitters. Activation products and carbon-14 were scaled using cobalt-60 activity, fission products were scaled using Cs-137 activity, and transuranics were scaled using Ce-144 activity. The inspector compared a list of scaling factors derived from the independent laboratory data to a computer print-out and verified that scaling factors had been accurately entered into computer programs.

The solid waste program was designed to ship solid wastes such as green-is-clean waste, dry active wastes, and resins and filters with dose rates less than 2 mSv/hr (200 mrem/hr) to a vendor facility for monitoring and/or volume reduction. Consequently, wastes in a final form ready for disposal were not generally stored on-site. Records showed that radioactive waste generation had steadily declined since the early 1970's. The radwaste staff attributed the decline to improved practices resulting from increased awareness, use of goals, and involvement of personnel from all major work groups in radwaste reduction efforts. Examples of improved practices included the following: 1) establishment of a green-is-clean program; 2) elimination of many disposable items such as plastics, tape, and tarpaulins; 3) increased use of re-useables such as cloth laundry bags, protective clothing, and rags; 4) elimination of some cloth filters and installation of a carbon bed water filtration system capable of being back-washed; 5) stopping the practice of pouring dirty mop water into floor drain systems; and 6) decreased use of chemicals and detergents that damage ion exchange resins.

The inspector noted that a 1997 goal had been set to reduce the volume of unprocessed low level radioactive waste to less than 3,320 ft^3; as of August 6, 1997, the actual volume generated was estimated to be 576 ft^3. In addition,

another 1997 goal had been set to reduce the volume of total buried low level waste to less than 100 ft^3; as of August 1997, the actual volume generated for burial was well below this goal.

The inspector performed tours of areas where radioactive materials were stored including the upper radwaste storage facility, the high integrity container (HIC) storage area, the radioactive material storage building, the contaminated storage building (CSB), and the drumming station. Overall volume of stored radioactive materials was low and materials were stored in a neat and orderly fashion. No wastes were stored in the HIC storage area; an inventory of materials was maintained at the upper radwaste storage area and materials were stored in a neat and orderly fashion; the radioactive material storage building was used for storage of materials primarily for outage use; and the drumming station was used to de-water resins, load high integrity containers for shipment, and for storing some wastes for eventual loading into HICs.

c. Conclusion

Based on this review, the inspector concluded the following:

- Procedures for the handling, packaging, classifying, and shipping radioactive materials/wastes were well maintained and of good quality, and copies of appropriate state and federal regulations and licenses for facilities were current and well maintained.

- Elements of the process control program appeared effectively managed as evidenced by clear procedural guidance, routine waste stream sampling, and validations of independent laboratory data.

- Efforts to reduce the generation of radwaste volume were effective, and radioactive materials were stored in a neat and orderly fashion.

R1.2 Radioactive Waste and Materials Shipping Program

a. Inspection Scope (86750)

A review was performed of the radioactive waste and materials shipping program. Information was gathered by a review of 1997 shipping records and discussions with cognizant personnel.

b. Observations and Findings

Shipping records were in file cabinets within the radwaste field office. Records were arranged in chronological order and were well documented and maintained. A review of shipments made in 1997 revealed that only 10 radioactive material shipments had been made including seven limited quantity shipments and three low specific activity (LSA) II shipments. Limited quantity shipments were excepted from a requirement to have shipping papers, but records included appropriate information

such as radiological survey data, isotopic activity, and type container. The LSA II shipments included appropriate information such as proper shipping name, name and activity of radionuclides, shippers certification, radiological survey data, and appropriate emergency response information. Overall, shipping records were well maintained, shipping class had been accurately identified, and no discrepancies were identified.

c. Conclusions

Based on this review the inspectors concluded that shipping records indicated that radioactive material shipments had been accurately classified and shipped in accordance with Department of Transportation (DOT) and NRC regulations.

R2 Status of RP&C Facilities and Equipment

R2.1 Review of Plant 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

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. Walkways and aisles were clear and free of debris, fresh paint had been applied in many areas, and lighting was generally good. Minor housekeeping discrepancies were identified in the boric acid evaporator room and a decontamination structure located within the contaminated storage building, and poor lighting was identified in the radioactive material storage area.

c. Conclusion

Based on this review, the inspector concluded that radiological boundaries were well maintained and clearly delineated, and conditions of housekeeping were generally good.

R5 Staff Training and Qualification in RP&C

R5.1 RP&C Staff Training

a. Inspection Scope (86750)

A review was performed of the training and qualifications of personnel involved with solid radioactive waste management and the transportation of radioactive wastes and materials. In addition, inspectors reviewed training materials developed

to inform personnel of their responsibilities regarding radiological boundaries. Information was gathered through a selected review of training records and course curriculum, and interviews with cognizant personnel.

b. Observations and Findings

Radwaste/Material Transportation

Training records indicated that individuals responsible for certifying the adequacy of radioactive waste shipments, including quality control personnel, had completed vendor training on the radwaste packaging and transportation regulations, and met qualification requirements. A selected review of training lesson plans indicated that course content was sufficient to satisfy training required by 49 CFR 172.700 Subpart H-Training (for hazmat employees). Training records also showed that health physics technicians had completed function specific training relative to the implementation of procedural guidance for the shipment and receipt of radioactive material/wastes. During interviews, health physics technicians assigned to the radwaste organization demonstrated very good knowledge of procedural guidance, methods for packaging and preparing radioactive shipments, and use of computer systems used to prepare and document radioactive material and waste shipments. Training records also showed that applicable industry events were included in continuing training provided to radwaste and health physics personnel.

Radiological Boundaries

Inspectors attended a mini-training (toolbox) session which communicated personnel responsibilities with regard to radiological boundaries. The training included a regulatory basis, industry events, causes of boundary problems, and boundary rules. The radiological controls department had been trained and a schedule had been established for training the entire plant staff. The training clearly articulated that personnel and equipment were required to enter/exit a contaminated area only at the step-off-pad; that only radiological protection (RP) technicians could remove material from controlled areas; and that reaching over a boundary was prohibited unless directed by an RP technician. The training was clear and succinct, and the training materials were informative.

c. Conclusions

Based on this review, the inspector concluded the following:

- Appropriate training on regulations and procedures pertaining to radioactive waste handling, processing, packaging, and shipping had been provided to the radwaste staff; staff members were qualified to perform tasks assigned within the radioactive material/waste management and transportation program; radwaste personnel demonstrated very good knowledge; and training programs were being effectively managed.

- Clear and informative training was developed to improve worker performance with regard to working in and around radiological boundaries.

R7 Quality Assurance in Radiological Protection and Chemistry Activities

R7.1 Quality Assurance Audits of Solid Radwaste Management and Transportation

a. Inspection Scope (86750)

A review was performed of quality assurance oversight of solid radwaste management and transportation of radioactive materials. Information was gathered by a review of a document entitled "Quality Assurance Program for Station Operation," audits performed in 1996 and 1997, action reports written within the last year related to radwaste activities, and discussions with radwaste and quality assurance personnel.

b. Observations and Findings

Requirements for the performance of an audit of the process control program and implementing procedures were contained in the document entitled "Quality Assurance Program for Station Operation, Rev. 23, dated December 13, 1996. The process control program and implementing procedures were listed as an audit topic area to be conducted on a 24 month cycle.

The inspector reviewed the following audits to evaluate quality assurance oversight of the process control program, solid radwaste management, and transportation of radioactive materials and wastes.

- AINT-1996-0005-NAB, "Audit of the Radiation Protection, Chemistry, Radioactive Waste and Environmental Monitoring," August 26, 1996.

- AINT-1996-0008-NAB, "Audit of the Radiation Protection, Chemistry, Radioactive Waste and Environmental Monitoring," August 11, 1997.

The audits noted good performance in housekeeping, minimizing the generation of wastes, and with documentation of stored materials. However, the audits were noticeably shallow in the evaluation of the process control program. The 1997 audit identified procedural guidance that was used to classify radwaste generated on-site and noted no problems with the process. However, based on the review of these audits, it appeared that a detailed review of radioactive waste stream sampling techniques, independent laboratory data, use of scaling factors, and use of computer programs to classify radwastes was not performed. The audits did not include a detailed review of radioactive waste stream sampling techniques, use of independent laboratory data, use of scaling factors, or use of computer programs to classify radwaste. This item is being treated as an unresolved item pending further review of audits and self assessments of the process control program (URI 50-244/97-09-02).

Based on a review of shipping records, the inspector noted that the quality control organization was actively involved in the preparation of radioactive shipments and as evidenced by documentation included in various shipping records.

Only one action report related to radioactive waste or material transportation had been written during the period from August 1996 to August 1997. The inspector reviewed ACTION Report No. 96-0813, "Empty Shipping Container Received in Excess of DOT Dose Rates," dated August 19, 1997. An empty shipping cask made out of depleted uranium was received at the Ginna site on August 19, 1996 at 1500 hours, with a maximum dose rate of 0.8 mrem/hr. The maximum allowable dose rate for a package shipped as empty class 7 (radioactive) materials packaging is 0.5 mrem/hr. The inspector noted that upon identification, the staff informed the shipper (the University of Missouri) of the finding, an action report was written to investigate the event, and the NRC resident inspectors were notified. This issue was also forwarded to NRC Region III for additional review and evaluation. In addition, corrective actions to prevent future occurrences, such as adding an overpack or changing the shipment classification for future shipments, were suggested to the shipper. The inspector noted that licensee actions were appropriate.

c. Conclusions

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

Overall quality assurance, quality control, and self assessment program oversight of the radioactive material/waste transportation program was effective. However, the review of the process control program documented in the 1996 and 1997 audits was noticeably shallow, and did not include detailed reviews in this area. This item is being treated as an unresolved item pending further review of audits and self assessments of the process control program **(URI 50-244/97-09-02)**.

R8 Miscellaneous RP&C Issues

R8.1 UFSAR Review

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspector reviewed Section 11.4, "Solid Radwaste Management." The inspector noted that some minor changes were made during the last update in December of 1996. During this review, no UFSAR discrepancies were identified in this area.

R8.2 Personal Protective Equipment Postings

The inspector noted that the industrial safety department had developed new signs to post requirements for personal protective equipment for entry into various areas. Some entrances to work areas had separate written notices for requirements for personnel protective equipment such as hard hats, safety glasses, and hearing protection. The new signs had colored pictures or "icons" that specified required

personal protective equipment, all on one sign. Additional clarifying information such as "hard hat required for overhead work" was also printed on the signs with the icon. The inspector concluded that the new signs were effective and removed some clutter from door postings.

R8.3 (Closed) Inspector Follow-up Item (IFI) 50-244/97-03-01

This item was opened because a material inventory had not been maintained for the upper radwaste storage facility. Although procedural guidance required the RP department to be notified when packages were moved to a storage location to ensure that radiological boundaries were properly maintained and that weekly surveys were performed, the inspector pointed out that maintenance of a current material inventory would help ensure control over radioactive material and radioactive wastes (e.g., to ensure appropriate radwaste packaging). During this inspection, the inspector was informed that in addition to existing radiological controls, a current inventory for the building was being maintained. The inspector reviewed this inventory sheet and noted that the inventory was reflective of the building contents. This item is closed.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Main Control Room Fire Suppression

a. Inspection Scope (64704)

The inspector reviewed the fire suppression systems and capabilities for the main control room, and the licensing basis and plant procedures related to control room fire suppression systems.

b. Observations and Findings

The control room at the Ginna Station is located directly adjacent to the turbine building and is part of the control building, which consists of the control room, the relay room, the air handling room, and the station battery rooms. The control room occupies a separate elevation above the other rooms in the control building, and represents a separate fire zone. The fire suppression systems and response procedures available for the control room are as follows:

The installed fire detection equipment consists of smoke and heat detectors only. These detectors activate audible alarms in the control room, but do not activate any automatic fire suppression equipment. The control room does not have an automatic fire suppression system (e.g., water sprinkler or halon system). Four portable carbon dioxide fire extinguishers are located inside the control room near the entrance/exit doors for use by the operators on any fire in the control room proper. The number and locations of these extinguishers were established in accordance with criteria of the National Fire Protection Association (NFPA) Code. One additional portable water extinguisher is located in the shift supervisor's office to deal with a small paper fire. The additional water extinguisher was added under

the Ginna Fire Hazards Analysis as additional protection. Abnormal procedure AP-CR.1, "Control Room Inaccessibility," directs operator actions for responding to fire, smoke, or toxic gases in the control room. In the event that control room evacuation is necessary due to a fire, emergency procedure ER-FIRE.1 directs operator actions for a remote plant shutdown outside the control room.

The licensee would dispatch the station fire brigade to combat any fire in the control room. The brigade is made up of the primary and secondary auxiliary operators and three security guards. The inspector noted that ER-FIRE.1 directs remote shutdown actions only for the licensed operators. No actions are directed for auxiliary operators, so that they can remain with the fire brigade for fire fighting. Designated fire equipment lockers are located just outside the control room's main entrance door, and contain protective clothing with five sets of self-contained breathing apparatus (SCBA) units for use by the fire brigade. Backup air tanks are also stored for each of the SCBAs. Two additional SCBA units are located inside the shift supervisor's office. These units were primarily intended for use by the on-shift radiological protection (RP) technician during a radiological emergency outside the control room; however, they would be available as backup equipment in the event of a fire. The portable fire extinguishers and protective equipment are all inspected and inventoried on a monthly basis. Fire response plan FRP-20.0, "Control Room," provides general fire fighting guidance to the fire brigade and to operators. The plan also contains important information related to fire fighting and detection equipment locations in the control room. Licensed operators receive annual simulator training on response to a control room fire; however that training is primarily aimed toward a control room evacuation followed by a remote plant shutdown.

The control room ventilation system (HVAC) has two "fire modes" of operation that are manually initiated by operator action. These modes reconfigure the HVAC system into partial recirculation flow lineups that can draw in fresh outside air or purge control room air to the outside. The fire modes were intended to maintain habitability in the control room for a small fire, and would also be useful for combatting or recovering from a large fire after operators evacuate the control room. The HVAC system is flow tested every 18 months, and includes system damper position verifications. However, the fire modes are not specifically tested on a periodic frequency, and the licensee agreed to investigate whether this should be accomplished.

The inspectors reviewed the licensee's procedures and the Updated Final Safety Analysis Report (UFSAR) sections for completeness and consistency with the existing fire suppression equipment for the control room. The abnormal and emergency response procedures contained adequate instructions for responding to a control room fire. FRP-20.0 contained an instruction that referenced a procedure that was cancelled in 1994. The licensee noted that approximately ten other FRPs also referenced cancelled procedures, and submitted procedure changes to make corrections. The inspectors noted that the UFSAR contained a reference to the four portable carbon dioxide extinguishers in the control room, but did not refer to the additional water extinguisher. The licensee considered that the additional extinguisher should be contained in the UFSAR and submitted a change include it.

Other UFSAR sections accurately referenced the numbers and locations of fire fighting equipment, and fire brigade staffing and response practices. However, the inspectors noted some UFSAR sections that contained information inconsistent with current plant equipment or practices as follows:

1) UFSAR Section 6.4, "Habitability Systems," incorrectly stated that the control room has "flame-sensing devices" installed; whereas, only smoke and heat detectors are installed. The licensee researched the basis for this statement and was not able to identify one. This statement will be removed from the UFSAR in a future revision.

2) UFSAR Section 9.5, "Fire Protection," stated that each fire brigade member receives 8 hours per year of hands-on training at the "Rochester Fire Academy." This facility is no longer in operation and the licensee conducts annual hands-on training in Oswego, NY at a facility owned by Niagara Mohawk. The licensee will revise this section as necessary.

3) UFSAR Section 9.5 stated that fire brigade training was based upon Section 27 of the NFPA Code-1975. However, Section 27 was deleted and superseded in 1986 by NFPA Section 600, which contained expanded requirements for fire brigade training programs. The licensee generated an ACTION Report to investigate whether appropriate actions were taken to address new program commitments that may have been necessary for Section 600.

c. Conclusions

The licensee's fire protection systems and capabilities were adequately supported by licensing basis documents, with the exception of minor inconsistencies. Operator actions for responding to a control room fire were adequately implemented by plant procedures and simulator training. The fire brigade was provided with appropriate procedures, protective equipment, and training to deal with a control room fire.

V. Management Meetings

X1 Exit Meeting Summary

After the inspection was concluded, the inspectors presented the results to members of licensee management on September 18, 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.

X3 Management Meeting Summary

X3.1 Division Director Visit

On August 6 and 7, 1997, Charles W. Hehl, Director, Division of Reactor Projects, Region I, conducted a tour of Ginna Station. Mr. Hehl observed systems and equipment at the Ginna facility and met with RG&E management and staff.

X3.2 Deputy Division Director Visit

On August 26 and 27, 1997, Richard V. Crlenjak, Deputy Director, Division of Reactor Projects, Region I, conducted a tour of Ginna Station. He was accompanied by Lawrence T. Doerflein, Chief, Reactor Projects Branch 1, Region I, and Guy S. Vissing, Project Manager, Office of Nuclear Reactor Regulation. The individuals observed systems and equipment at the Ginna facility and met with RG&E management and staff.

L2 Review of UFSAR Commitments

A review of the Updated Final Safety Analysis Report (UFSAR) was conducted that compared plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR that related to the areas inspected. Several inconsistencies were noted between the UFSAR and the licensee's fire protection program as described in section F2.1 above.

ATTACHMENT I

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Flynn	Primary Systems Engineering Manager
C. Forkell	Electrical Systems Engineering Manager
G. Graus	Electrical/I&C Maintenance Manager
A. Harhay	Radiological Protection & Chemistry Manager
D. Filion	Radiochemist
J. Hotchkiss	Mechanical Maintenance Manager
G. Joss	Results and Test Supervisor
R. Jaquin	Nuclear Safety and Licensing
R. Marchionda	Production Superintendent
F. Mis	Principle Health Physicist
P. Polfleit	Emergency Preparedness Manager
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 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 83750:	Occupational Radiation Exposure
IP 86750:	Solid Radioactive Waste Management and Transportation of Radioactive Materials.
IP 92903:	Follow-up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

IFI 50-244/97-09-01	In-plant JPM Cuing Techniques
URI 50-244/97-09-02	QA Audits of Radwaste program

Closed

IFI 50-244/97-03-01	Upper Storage Facility Activity Limits
URI 50-244/96-07-01	Containment Spray Test Line Leakage
URI 50-244/96-07-02	Minimum AFW Flow Requirements for Accident Conditions
LER 96-009/Rev 2	Leak Outside Containment, Due to Weld Defect, Results in Leak Rate Greater Than Program Limit
LER 97-003	Bistable Instrument Setpoint (Plus Instrument Uncertainty) Could Exceed Allowable Value, Causes a Condition Prohibited by Plant Technical Specifications

Discussed

URI 50-244/96-06-04	Thermal Performance Test Program for Service Water Heat Exchangers

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
AO	Auxiliary Operator
CCW	Component Cooling Water
CSB	Contaminated Storage Building
CFR	Code of Federal Regulations
CV	Check Valve
DAW	Dry Active Waste
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESF	Engineered Safety Feature
FME	Foreign Material Exclusion
FRP	Fire Response Plan
gpm	gallons per minute
HEPA	High Efficiency Particulate Air
HIC	High Integrity Container

LIST OF ACRONYMS USED (continued)

HVAC	Heating, Ventilation, and Cooling
IFI	Inspector Follow-up Item
IR	Inspection Report
ITS	Improved Technical Specifications
JPM	Job Performance Measure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LSA	low specific activity
LWP	Liquid Waste Processing
MOV	Motor-Operated Valve
MDAFW	Motor-Driven Auxiliary Feedwater
MRPI	Multiplexing Rod Position Indication System
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NS&L	Nuclear Safety and Licensing
P&ID	Piping and Instrumentation Drawing
PORC	Plant Operations Review Committee
PRT	Pressurizer Relief Tank
PT	Periodic Test
psig	pounds per square inch gage
QA	Quality Assurance
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG&E	Rochester Gas and Electric Corporation
RHR	Residual Heat Removal
RP	Radiological Protection
SCBA	Self Contained Breathing Apparatus
SFP	Spent Fuel Pool
SG	Steam Generator
SI	Safety Injection
SRO	Senior Reactor Operator
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order