November 6, 2003

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT- NRC INTEGRATED INSPECTION REPORT 05000244/2003006

Dear Dr. Mecredy:

On September 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your R. E. Ginna facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 16, 2003, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified four issues of very low safety significance (Green). Two of these issues were determined to involve a violation of NRC requirements. However, because of their very low safety significance, and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in Section 4OA7 of this report. If you deny the non-cited violations noted in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Ginna facility.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction (TI) 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the order. Phase 1 of TI 2515/148 was completed at all commercial power nuclear power plants during calender year 2002 and the remaining inspection activities for Ginna were completed in August 2003. The NRC will continue to monitor overall safeguards and security controls at Ginna.

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

Sincerely,

/RA/

James M. Trapp, Chief
Projects Branch 1
Division of Reactor Projects

Docket No. 50-244
License No. DPR-18

Enclosure: Inspection Report 05000244/2003006
w/ Attachment: Supplemental Information

cc w/encl:
J. Laurito, President, Rochester Gas and Electric
P. Eddy, Electric Division, Department of Public Service, State of New York
C. Donaldson, Esquire, State of New York, Department of Law
N. Reynolds, Esquire, Winston & Strawn
P. Smith, Acting President, New York State Energy Research and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority
D. Stenger, Ballard, Spahr, Andrews and Ingersoll. LLP
T. Wideman, Director, Wayne County Emergency Management Office
M. Meisenzahl, Administrator, Monroe County, Office of Emergency Preparedness
T. Judson, Central New York Citizens Awareness Network

Distribution w/encl:
H. Miller, RA/J. Wiggins, DRA (1)
J. Jolicoeur, RI EDO Coordinator
R. Laufer, NRR
R. Clark, PM, NRR
P. Milano, PM, NRR (Backup)
K. Kolaczyk, SRI Ginna
M. Marshfield, RI Ginna
J. Trapp, DRP
N. Perry, DRP
Region I Docket Room (with concurrences)

DOCUMENT NAME: G:\BRANCH1\Ginna\Reports\GIN0306.wpd
After declaring this document “An Official Agency Record” it **will/will not** be released to the Public.**To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP				
NAME	KKolazcyk/DF for		JTrapp/MS for				
DATE	11/5/03		11/5/03				

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-244
License No:	DPR-18
Report No:	05000244/2003006
Licensee:	Rochester Gas and Electric Corporation (RG&E)
Facility:	R. E. Ginna Nuclear Power Plant
Location:	1503 Lake Road Ontario, New York 14519
Dates:	June 29, 2003 - September 27, 2003
Inspectors:	K. Kolaczyk, Senior Resident Inspector M. Marshfield, Resident Inspector G. Hunegs, Senior Resident Inspector, Nine Mile Point D. Dempsey, Resident Inspector, Fitzpatrick S. Dennis, Resident Inspector, Oyster Creek J. D'Antonio, Operations Engineer G. Bowman, Reactor Inspector P. Frechette, Physical Security Inspector P. Harris, Operations Engineer J. Jang, Senior Health Physicist F. J. Laughlin, Emergency Preparedness Specialist D. Merzke, Reactor Inspector T. Moslak, Health Physicist N. Perry, Senior Project Engineer J. Schoppy, Senior Reactor Inspector D. Silk, Senior Emergency Preparedness Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000244/2003-006; 06/29/2003 - 09/27/2003; R. E. Ginna Nuclear Power Plant; Maintenance Risk Assessment and Emergent Work, Personnel Performance During Non-routine Plant Evolutions, Post Maintenance Testing, Other Activities.

The report covered a 3-month period of inspection by resident inspectors and announced inspections by regional specialists. This inspection identified two Green non-cited violations (NCVs) and two Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified that RG&E did not have compensatory measures in place, to prevent the air temperature in the relay room from exceeding the maximum values described in the plant Updated Final Safety Analysis Report (UFSAR). High air temperatures in the relay room would degrade the performance of safety-related components located in that room.

This finding is greater than minor because it is associated with the procedure quality attribute of the mitigating systems cornerstone and adversely affects the cornerstone objective because high temperatures in the room would not assure the reliable operation of systems needed to respond to an initiating event. This finding is of very low safety significance since the excessive temperatures would not be reached for several hours, which affords time for the operators to take action(s) to mitigate the temperature rise. (Section 1R13)

Green. A self-revealing non-cited violation of Technical Specification 5.4.1.a was identified due to the operating crew not correctly implementing procedures ES-0.1 "Reactor Trip Response." This resulted in a period of inoperability for the "B" motor driven auxiliary feedwater pump.

This finding is greater than minor because it involved a human performance error which resulted in reduced capability of a mitigating system, specifically auxiliary feedwater. This finding, which is under the mitigating systems cornerstone, is of very low safety significance because it was an actual loss of safety function of a single train or multi-train system for approximately three days, a time less than the Technical Specification allowed outage time of seven days.

A contributing cause of this finding is related to the Human Performance cross-cutting area. Inadequate placekeeping in the procedure by the operating crew resulted in the omission of the step to shutdown the "B" motor driven auxiliary feedwater pump. (Section 1R14)

Green. The RG&E vendor manual control program was inadequate in that it did not ensure maintenance personnel were provided with the information needed to properly rebuild the lubricating oil circulation pump for the "A" motor driven auxiliary feedwater pump. As a result, the pump was not properly assembled during maintenance activities.

This finding is greater than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective. The lubricating oil circulation pump must be operable to ensure the MDAFW pump can meet its design functions of mitigating an event. This finding was determined to be of very low safety significance (Green) using Phase 2 of the Significance Determination Process (SDP) under the Mitigating Systems cornerstone. (Section 1R19)

Cornerstone: Barrier Integrity

Green. While observing maintenance activities on the spent fuel pool system charcoal filtration system, the inspectors identified that contrary to requirements in the applicable maintenance procedure, RG&E personnel were working on the system when spent fuel was being moved in the spent fuel pool. The failure to correctly implement the maintenance procedure was a violation of Technical Specification (TS) 5.4.1.a which states, in part, that procedures shall be established, implemented and maintained.

This finding is greater than minor because it is similar to example 2.h of NRC manual Chapter 0612, Appendix E, "Power Reactor Inspection Reports" where multiple examples of personnel failing to follow procedures have occurred. This finding was of low safety significance since the fuel assemblies that were being moved at the time that maintenance personnel were realigning the ventilation system were not recently irradiated assemblies. Therefore, in the event they were damaged, a significant offsite release of unfiltered radioactive particles would not have occurred. (Section 1R19)

B. Licensee-Identified Violations

Violations of very low safety significance which were identified by RG&E were reviewed by the inspector. Corrective actions taken or planned by RG&E appeared reasonable. These violations are summarized in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Ginna began the period at full power. On August 14, 2003, a protective reactor trip occurred as a result of load fluctuations on the offsite electrical grid. Following the completion of maintenance activities, the plant was restarted and connected to the grid on August 17. The plant reached full power on August 19, and remained there until September 1, when a coastdown period was entered due to fuel depletion. On September 15, the plant was taken off-line to commence a planned refueling outage.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed what actions RG&E took to prepare for the arrival of Hurricane Isabel on September 19, 2003. Included in the review was an examination of the Ginna severe weather procedure ER-SC-1, “Adverse Weather Plan,” and a review of upcoming outage work activities. This activity counted for one sample.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

Recent industry events involving Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 at other plants throughout the industry prompted RG&E to take the preemptive measure of replacing the reactor vessel closure head (RVCH) during their Fall 2003 refueling outage. The design of the new RVCH is similar to the existing RVCH except for the replacement of the Alloy 600 penetration tube material and Alloy 600 weld material with a new and improved PWSCC resistant material (Alloy 690) and several other minor improvements.

The new RVCH was made as a single forging and clad with stainless steel on the inside in Japan, then machined and fabricated with welded control rod drive mechanism (CRDM) guide tubes, and hydro-pressure tested in Canada. The CRDMs were manufactured in France and shipped to the Ginna site where they were attached and seal welded to the CRDM guide tube adapters prior to the outage. Early in the 2003 refueling outage, RG&E moved the new RVCH into containment to replace the existing RVCH.

The inspectors verified that RG&E performed the RVCH-related design changes and modifications to structures, systems, and components (SSCs) described in the Updated Final Safety Analysis Report (UFSAR) in accordance with 10CFR50.59. The inspectors reviewed RG&E’s evaluations of applicability determination and screening questions for each design change or modification to determine, for each change, whether a 10CFR 50.59 had been screened out or performed, and the justification for each.

Specifically, the inspectors reviewed Plant Change Record (PCR) 2001-0042, which included a review of the function of each changed component, the change description and scope, and the 10CFR 50.59 evaluations for the following items which accounted for eight samples:

- RVCH replacement
- CRDM replacements with improved drives
- Improved core exit thermocouple nozzle assemblies (CETNAs)
- CRDM cooling coil shroud improvement
- Removal of four unused part length CRDMs
- Replacement of the RVCH insulation inside the cooling shroud
- Removal of the spare Core Exit Thermocouple column and head penetration
- Relocation of the RVCH vent and separation from the reactor vessel level indication system (RVLIS)

The inspectors reviewed Framatome Advanced Nuclear Power's (ANP) certified design report (6 CS 1075) for the CRDM pressure housing assembly. This report provided a computerized analysis of the CRDM to satisfy the applicable requirements of the ASME Boiler and Pressure Vessel Code Section III (1995 edition including addenda through 1996). In addition, the inspectors verified that RG&E engineering performed a stress analysis for the new RVCH and considered the as-built dimensions in this analysis.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope (71111.04Q)

Partial System Walkdowns. The inspectors completed seven samples which were partial walkdowns of the following system/trains:

- “A” Battery Room
- “B” Battery Room
- “A” Diesel Generator
- “B” Diesel Generator
- “A” Residual Hear Removal Train
- “B” Auxiliary Feed Water
- Service Water System

These inspections reviewed alignment of system valves and electrical circuit breakers to ensure proper in-service or standby configurations described in plant procedures and drawings. During the walkdowns, the inspectors also evaluated material conditions and general housekeeping of the systems and adjacent spaces. The condition of the “A” and “B” diesel generators were examined, when their complimentary diesel generator was out of service for maintenance. Both batteries were selected for a walkdown to ensure scaffolding that was being installed in both rooms in support of a modification to the room smoke detectors, did not affect operability of the batteries. The “A” RHR system was selected for review since the “B” train was out of service for planned maintenance. The “B” auxiliary feedwater train was walked down following corrective maintenance to the pump’s air-operated recirculation valve. The service water system was selected due to its risk significance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors completed nine samples by performing walkdowns of the following fire areas to determine if there was adequate control of transient combustibles and ignition sources. The material condition of fire protection systems, equipment and features, and the material condition of fire barriers were also inspected against industry standards. In addition, the passive fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. The following plant areas were inspected:

- Screenhouse Operating and Basement Floors
- Intermediate Building - Fan Deck
- Cable Tunnel
- Containment
- Control Room
- Diesel Generator Room “A”
- Diesel Generator Room “B”
- Battery Room “A”
- Battery Room “B”

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspector reviewed RG&E’s periodic maintenance, testing, and inspection records for the following safety-related heat exchangers to determine if RG&E had reasonable assurance that the heat transfer capability for each heat exchanger would remain capable of meeting its design heat removal requirements during plant operations. Two inspection samples were completed as a result of this activity.

- “B” Containment Recirculation System Fan Cooler
- “B” Containment Recirculation Fan Cooler Motor

As part of the review, the inspector discussed the test results with the system engineers for the service water and containment recirculation fan cooler systems. The inspector also reviewed the service water system program document, “Service Water System

Reliability Optimization Program," and the applicable sections of the plant Updated Final Safety Analysis Report (UFSAR).

b. Findings

No findings of significance were identified

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors selected a sample of nondestructive examination (NDE) activities for review. This sample included radiographic tests (RT), an ultrasonic test (UT), a liquid penetrant test (PT), and a visual exam (VT). For each of the NDE activities reviewed, the inspectors verified that the examination was conducted in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and that any indications were properly dispositioned.

The inspectors reviewed RT results on risk significant welds in the "High Energy Line Break" program (including pipe-to-reducer welds and valve-to-reducer welds covered under report 03GRT054M). The inspectors also reviewed the licensee's plan to use RT in place of UT on pressurizer spray and relief valve nozzles, as well as plans to use radiographic methods to size flaws. This review included inspection of the Electric Power Research Institute (EPRI) mock-ups used to qualify the examination technique, as well as calculations and error analysis for the flaw sizing approach.

The inspectors observed a PT conducted on a weld inside the containment in the residual heat removal (RHR) system suction line from the reactor hot leg. The inspectors reviewed the qualification records of the personnel performing the examination and verified that procedural controls were adequate. The inspectors independently assessed the results of the test for comparison with RG&E.

The inspectors observed an UT conducted on the weld described above. The inspectors observed calibration and preparation of the UT equipment and verified the evolution was conducted in accordance with an approved and acceptable procedure. The inspectors independently assessed the results of the test for comparison with RG&E.

Visual examination results were reviewed as part of the inspection of RG&E's activities performed in response to Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity." These activities were inspected against the requirements of Temporary Instruction (TI) 2515/152. A description of the scope and results of this inspection is found in section 4OA5 as specified by the TI.

The inspectors verified RG&E has been identifying inservice inspection (ISI) related problems at an appropriate threshold and properly entering them in the corrective action program. The inspectors also reviewed a sample of corrective action documents generated as a result of ISI activities to ensure problems were resolved in a manner commensurate with risk.

RG&E did not conduct steam generator tube inspections during this outage. Additionally, the inspectors determined that RG&E's ISI program corrected recordable indications when found, rather than accepting them for continued service. Therefore, the inspectors did not review recordable indications from the previous outage that were left in service as discussed in Inspection Procedure 71111.08.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspector observed portions of a licensed operator training scenario conducted on July 7, 2003. As a result, one sample was completed. The training scenario was ES1213-03 Rev.9, "Large Break Loss of Coolant Accident." The inspector reviewed the critical tasks associated with the evaluation, observed the operators' performance during the exercise, and observed the post-evaluation critique. The inspector also reviewed and verified compliance with Ginna procedure OTG-2.2, "Simulator Examination Instructions."

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed how RG&E used the maintenance rule to address performance-related issues associated with the charging pumps in the chemical and volume control system, and relief valves in the condensate and feedwater systems. Specific areas reviewed included scoping, performance criteria/ goal monitoring, and problem classification. Two samples were completed from the following systems:

- Chemical and Volume Control System 07
- Condensate System 84A

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

On six occasions, the inspectors evaluated the effectiveness of RG&E's maintenance risk assessments required by paragraph a(4) of 10 CFR 50.65. This inspection included discussions with control room operators and scheduling department personnel regarding the use of RG&E's online risk monitoring software. The inspectors reviewed equipment tracking documentation, daily work schedules, and performed plant tours to gain reasonable assurance that actual plant configuration matched the assessed configuration. Additionally, the inspectors verified that RG&E's risk management actions, for both planned and/ or emergent work, were consistent with those described in procedure IP-PSH-2, "Integrated Work Schedule Risk Management." Risk assessments for the following out of service systems, structures, and/or components were reviewed. The following six samples were completed:

- Unplanned troubleshooting activities performed on August 4, 2003, for the diesel generator tie-in breaker to electrical bus 16.
- Unplanned maintenance on the relay room air conditioning systems performed on July 14, 2003, to address a high temperature condition in the relay room.
- Unplanned troubleshooting activities performed on July 21, 2003, on the undervoltage protection circuitry for electrical Bus 16 to diagnose the reason(s) why an electrical fuse in the circuitry suddenly failed.
- Troubleshooting activities performed on the control rod drive system on August 16 and 17 following the August 14, 2003, reactor plant trip.
- Reviewed planned maintenance on Instrument Air (IA) Compressor "B" while IA Compressor "C" was in an emergency use only status. This maintenance was canceled when operations noted a top level "Orange" risk for the system.
- Corrective actions implemented on September 4, 2003, to address a break in a water main that supplied water to the plant fire main system.

b. Findings

Introduction. The inspectors identified that RG&E did not have compensatory measures in place to prevent the air temperature in the relay room from exceeding the maximum values described in the plant Updated Final Safety Analysis Report (UFSAR). High air temperatures in the relay room would degrade the performance of safety-related components located in that room.

Description. The relay room is located beneath the Ginna control room; it is cooled by two non-safety-related air conditioning systems that would deenergize in the event a loss of offsite power occurred. In addition to containing cables for the instrumentation in the control room, the relay room contains instrumentation and control equipment for the plant process computer, control rods, advanced digital feedwater control system, and undervoltage relay protection cabinets. For environmental qualification purposes, the relay room is classified as a mild environment. Table 3.11-1 of the Ginna UFSAR indicates the room temperature will not exceed 104°F under accident conditions. Chapter 3.11.3.5 of the Ginna UFSAR states that operators may have to use portable air conditioners and natural circulation methods to keep the room temperature within design assumptions.

The inspector noted that an October 5, 1990, engineering study of the relay room ventilation system concluded the room air temperature could reach 130° F sixty-five hours after cooling was lost to the room if compensatory measures were not implemented. This conclusion was confirmed in a January 13, 1994, re-analysis of the room ventilation system.

Although the Ginna UFSAR and engineering analysis indicated compensatory measures would have to be implemented to cool the relay room in the event the room coolers became deenergized, the inspector determined there were no plant procedures that described what action(s) should be taken. This observation was discussed with RG&E engineering personnel who initiated Action Report 2003-1745, "No Procedure Guidance for Loss of Relay Room Cooling."

Analysis. The performance deficiency in this event is a failure to provide procedures for maintaining relay room temperatures below those assumed in the UFSAR in the event of a loss of all air conditioning systems. This finding is greater than minor, because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and adversely affects the cornerstone objective because high temperatures in the room would not assure the reliable operation of systems needed to respond to an initiating event. In accordance with Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a SDP Phase 1 screening and determined that the finding is of very low safety significance (Green). The SDP process screens directly to (Green) since the excessive temperatures would not be reached for several hours, which affords time for the operators to take action(s) to mitigate the temperature rise. **(FIN 05000244/2003006-01, No Procedure Guidance for Loss of Relay Room Cooling)**

Enforcement. No violation of NRC requirements occurred.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

a. Inspection Scope

The inspector reviewed operator response to a reactor trip on August 14, 2003, during which the "B" auxiliary feedwater pump was damaged due to operator error identified by RG&E. The inspector interviewed the crew and reviewed the following documentation for this one sample:

- Results from 40 evaluated simulator scenarios, including annual operating examinations and "as found" training scenarios.
- Facility guidance on Emergency Operating Procedure usage and placekeeping.
- Lesson plans for the auxiliary feedwater system.
- Action Reports involving operator errors and/or training.

At approximately 4:11 p.m. on the afternoon of August 14, 2003, the northeast US power grid experienced a significant instability which resulted in large power output variations on the main generator. The plant systems attempted to compensate by introducing a turbine runback. The combination of reduced power transfer from the reactor coolant system and the turbine runback combined to generate a valid over temperature differential temperature (OTΔT) reactor trip followed by an associated turbine trip. This sequence of events took approximately eight seconds and included lifting of both Pressurizer Relief Valves. The operators entered E-0, "Reactor Trip or Safety Injection," and subsequently transitioned to ES-0.1, "Reactor Trip Response," for a normal reactor trip. The instability also caused both reactor coolant pumps to trip on underfrequency. At 4:49 p.m. RG&E declared an Unusual Event based on instability of the off-site power source lasting longer than 15 minutes. Additional complications during the trip resulted in advanced digital feedwater control system (ADFCS) failing to “manual” and subsequently overfeeding the steam generators. As a result, the main steam isolation valves were shut to regain steam generator level control and the reactor plant was stabilized in Hot Standby in natural circulation with the atmospheric relief valves in use to remove decay heat. Several hours into the event, a subsequent operator error while using ES-0.1 resulted in damage to the “B” motor driven auxiliary feedwater (MDAFW) pump. The emergency diesel generators (EDG)s were started manually and vital loads were placed on the EDGs because of the instability of the offsite power supply. Non-vital loads were not lost and remained on off-site power throughout the event.

The inspectors were on-site for the event and responded to the control room. While in the control room, the inspectors verified operators were adhering to procedures, and were taking appropriate actions to mitigate the event. Inspectors walked down control room panels to ensure plant temperatures and pressures were within expected parameters.

During the extended period of plant recovery, the inspectors monitored the restart of forced coolant flow, restoration of the main condenser as an effective heat sink, repairs to the “B” MDAFW pump, and securing of the EDGs. Plant response and actions were reviewed for compliance with procedures and proper system response. The plant computer sequence of events printout was reviewed and compared to plant data collected from plant logs. The reactor was restarted and the generator placed on the grid August 17, 2003, at 8:38 p.m. The inspectors maintained 24-hour coverage from the time of the grid instability until the reactor was returned to power.

b. Findings

Introduction. A Green self-revealing violation of very low safety significance was identified when plant operators did not shutdown the "B" auxiliary feedwater pump as required by ES-0.1, "Reactor Trip Response,” prior to opening the discharge header crossover valves. This overheated and damaged the “B” MDAFW pump.

Description. While responding to a reactor trip on August 14, 2003, the crew implemented ES-0.1, "Reactor Trip Response.” Step 19 of this procedure directs the crew to establish normal AFW pump shutdown alignment by stopping one of two MDAFW pumps, then opening the discharge header crossover valves. Due to inadequate placekeeping, the operating crew omitted the step to shut down one MDAFW pump. With both pumps running and the crossover valves open, the pump with higher discharge pressure ("A") deadheaded the other pump ("B") resulting in damage from overheating. This error resulted from inadequate placekeeping in the procedure. The facility has no formal placekeeping guidance, but rather allows the operators to utilize any method that suits them.

Analysis. The performance deficiency associated with this event is failure to perform a step in an Emergency Operating Procedure (EOP), which resulted in damage to the "B" auxiliary feedwater pump. This finding is greater than minor because it involved a human performance error which affected the Mitigating System Cornerstone in that the secondary heat removal capability of the auxiliary feedwater system was reduced. In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance

Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted an SDP Phase 1 screening and determined that the finding was of very low safety significance (Green). Specifically, the finding involved an actual loss of safety function of a single train of a multi train system for approximately three days, a time less than the Technical Specification allowed outage time of seven days; and the finding did not involve the total loss of a safety function that contributes to external event initiated accident sequences.

A contributing cause of this finding is related to the Human Performance cross-cutting area. Inadequate placekeeping in the procedure by the operating crew resulted in the omission of the step to shutdown the “B” motor driven auxiliary feedwater pump.

Enforcement. Technical Specification 5.4.1 (a) requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A. Item 2.c of this Regulatory Guide requires a procedure for Recovery from Reactor Trip. Contrary to the above, the operating crew did not correctly implement step 19.d of ES-0.1 "Reactor Trip Response" in that they did not stop one motor driven auxiliary feedwater pump before opening the auxiliary feedwater discharge crossover valves in step 19.e. The Ginna corrective action document addressing this condition is AR 2003-1821, “B AFW Pump Damage Human Performance Issues.”

This failure to follow procedure requirements is of very low safety significance (Green) and has been entered into the corrective action program; this violation is being treated as a non cited violation (NCV), consistent with Section VI.A of the NRC enforcement policy: **NCV 05000244/2003006-02, Operators Did Not Shutdown “B” MDAFW pump per ES-0.1.**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following five samples of operability evaluations to determine if system operability was properly justified by RG&E:

- Action Report (AR) 2003-0309, “AOV 966C Exceeds Administrative Leakage Limit.”
- Action Report (AR) 2003-1550, “Service Water Leak Downstream of V-4619."
- Action Report (AR) 2003-1933, “Spent Fuel Pool Recirc Pump A Check Valve,”
- Action Report (AR) 2003-1720, “Control Room Roof Leakage,”
- Action Report (AR) 2003-2024, “Pump Minimum Flow Output Not Obtained, B Charging Pump”

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

1. Control Rod Drive Mechanism Modification

a. Inspection Scope

RG&E mounted 29 full length Control Rod Drive Mechanism (CRDMs) to the new reactor vessel closure head (RVCH) penetration housing adapters (see Section 1R02). The CRDMs function as an extension of the primary pressure boundary and consist of the latch housing, the rod travel housing, and the rod travel housing cap. RG&E planned to replace the existing model L-106 drives with an equivalent L-106 drive (model L-106A). The L-106A drive is similar to the L-106 drive except for incorporating a seal welded joint at the rod travel housing to latch assembly housing joint which eliminates the gasket and bolted joint of the L-106 model.

On August 26 - 28, 2003, the inspectors directly observed Framatome's CRDM installation activities. The inspectors also reviewed weld records and corrective action documents associated with the following CRDM installation activities:

- CRDM rigging and transport
- CRDM-adapter fit-up before welding
- CRDM-adapter seal welds
- CRDM-adapter seal weld liquid penetrant examinations

The inspectors reviewed additional quality records and procedures to verify that Framatome and RG&E performed and documented their CRDM work in accordance with requirements. The inspectors verified that Framatome and RG&E established and implemented appropriate foreign material exclusion controls during these activities. The inspectors also observed in-process quality assurance (QA) oversight activities to ensure that RG&E applied adequate oversight. This inspection activity accounted for one sample.

b. Findings

No findings of significance were identified.

2. Head Assembly Upgrade Package

a. Inspection Scope

The inspectors observed portions of and reviewed documents supporting installation of the Head Area Upgrade Package (HAUP), RVCH vent nozzles, and CETNAs (see Section 1R02). The inspector reviewed the documents, including the material used for fabrication, to ensure that Framatome and RG&E performed these installation activities in accordance with design drawings and quality requirements.

b. Findings

No findings of significance were identified.

3. Battery Room Fire Detection Modification

a. Inspection Scope

The inspectors reviewed plans for and observed the completion of Plant Change Record (PCR) No. 2000-0048, "Smoke Detection Upgrades." This plant modification was conducted as a result of design analysis following a fire detection system self-evaluation on September 9, 2000. The modification was conducted to improve smoke detection capability in the battery rooms, above the condensate booster pumps and above busses 12A and 12B. The previous installation included two detectors on each beam in the battery rooms. This previous configuration was contrary to current National Fire Protection Association (NFPA) guidance (the 2002 version of NFPA-72 was used to plan this job) which requires a detector in each beam pocket to ensure early detection of smoke in an overhead area. Since RG&E was not formally committed to the 2002 version of the NFPA code, the upgrade was a voluntary initiative to improve detection capability and detector surveillance since the upgraded detectors would also have a continuous self-checking feature. Satellite stations, which control the fire sensing system operations, were also upgraded or replaced as a part of the modification to improve the overall fire alarm system response and improve maintainability.

The inspectors reviewed the modification preparations, planning and execution. Fire wall penetrations, extensive scaffolding installations in the battery rooms, fire watch posting requirements and subsequent restorations of systems were observed during the conduct of the modification. The safety significant portions of the modification in the battery rooms and the satellite stations were walked down during the maintenance and on completion of the installation. Post maintenance testing was evaluated for adequacy. This inspection activity accounted for one sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance tests for the following work orders (WO) to verify that RG&E appropriately demonstrated the components' ability to perform their intended safety function. Six samples were completed.

- WO 20203074, "EDG Fuel Oil Strainer Inspection"
- WO 20203224, "Replace Mechanical Seals 'A' CCW Pump"
- WO 20203229, "Spent Fuel and Decon Pit Exhaust System"
- WO 20203485, "Inspect CV-5941A, EDG A Starting Air Compressor and Discharge Valve"
- WO 20300119, "Charging Pump C Varidrive Overhaul"
- WO 20301772, "Inspect 4L Cylinder 'B' Diesel Generator"

b. Findings

Finding 1 - Auxiliary Feedwater Pump Lubricating Oil Circulation Pump Failure

Introduction. The RG&E vendor manual control program was inadequate in that it did not ensure maintenance personnel were provided with the information that was needed to properly rebuild the lubricating oil circulation pump for the "A" motor driven auxiliary feedwater pump. As a result, the pump was not properly assembled during maintenance activities.

Description. On August 12, 2003, during a quarterly surveillance test, an electrically driven gear pump, which circulates lubricating oil for the "A" MDAFW pump reduction gear, tripped. No obvious cause(s) of the failure were found, but due to previous problems with the pump's electrical thermal overload protective devices prematurely tripping, they were replaced as a precautionary measure. The pump was restored to service, and operated without incident for three days during the August 14, 2003, power grid problem until the main feedwater system was placed into service on August 17.

Unable to diagnose a root cause(s) for the August 12, 2003, pump failure, the Plant Operations Review Committee (PORC) recommended that the "A" MDAFW pump be tested more frequently. On September 5, during the first augmented test, the pump was started and lube oil pressure reached normal conditions. After a few seconds, the discharge pressure for the lube oil pump, as indicated on the locally mounted gauge, dropped to zero. Operators then stopped the MDAFW pump and commenced troubleshooting. During subsequent runs of the lube oil pump, the failure could not be reproduced. Nevertheless, during examination of the pump, maintenance personnel identified that the pump's shaft axial movement allowance was not set within the specifications established by the pump vendor when the pump was rebuilt in December 2002. As a result, there was insufficient clearance between the pump's internal mechanisms, which could cause the pump to bind. Other problems RG&E noted with the lubricating oil system included a filter located downstream of the pump that was installed backwards and the system return pipe had been placed too far into the reduction gear sump, partially blocking return flow.

The pump was not set to the specifications established by the vendor, because during the December 2002 pump rebuild, RG&E maintenance personnel did not have the requisite vendor manual which provided the necessary specifications. One possible reason for not having the necessary vendor manual was the procedure governing update of vendor manuals IP-RDM-2, "Vendor Technical Document Control and Change Requests" did not require RG&E to ask their vendors if RG&E had all the technical information that they needed for their components.

On September 6, 2003, the pump's axial clearance was restored to within the vendor recommended values, tested, and declared operable. The lube oil pump for the "B" MDAFW was then examined. No discrepancies were noted.

Analysis. The performance deficiency in this event is that RG&E did not ensure maintenance personnel had sufficient information to maintain the lubricating oil pump. As a result, the pump was not properly aligned. This finding is greater than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective. The lubricating oil circulation pump must be operable to ensure the MDAFW pump can meet its design functions of mitigating an event.

To assess the significance of this condition, the inspectors performed a Phase 2 Significance Determination Process (SDP) analysis of the MDAFW pump failure using the reactor safety SDP. Given that the "A" MDAFW pump had successfully operated when called upon on August 14, the inspectors assumed the "A" MDAFW had been out of service for greater than three days but less than 30. The inspectors also assumed the remaining complimentary mitigating systems including the "B" MDAFW pump and the standby AFW pumps were operable.

The analysis concluded that the most limiting risk significant sequence was a loss of offsite power event followed by failure of the turbine driven, motor driven, and standby auxiliary feedwater pumps, with a failure of control room operators to commence a feed and bleed of the reactor coolant system. Given the number of failures that had to occur before core damage event would occur, the SDP concluded that the September 5, 2003, failure of the "A" MDAFW pump was an issue of very low significance, or (Green). **(FIN 05000244/2003003-03 Vendor Manual Control Program was Inadequate)**

Enforcement. RG&E's failure to provide maintenance personnel with adequate instructions for rebuilding the pump is a licensee-identified violation that is discussed in section 4OA7 of this report.

Finding 2 - Spent Fuel Pool Ventilation System Maintenance

Introduction. While observing maintenance activities on the spent fuel pool system charcoal filtration system, the inspectors identified that contrary to requirements in the applicable maintenance procedure, RG&E personnel were working on the system when spent fuel was being moved in the spent fuel pool. The failure to correctly implement the maintenance procedure was a violation of Technical Specification (TS) 5.4.1 which states, in part, that procedures shall be established, implemented, and maintained.

Description. On August 4, 2003, while watching RG&E personnel align the spent fuel pool ventilation system in preparation for testing of the charcoal filtration media, the inspectors noted that contrary to the requirements contained in the applicable maintenance procedure M-7.9, “Spent Fuel and Decon Pit Exhaust System Plenum Installation/Removal of Media Filters/Blanking Plates/Frames,” the system alignment was being performed when fuel was being moved in the spent fuel pool. Upon discovering the procedure noncompliance, the inspector informed a health physics (HP) technician who was overseeing the work activity. The HP technician promptly informed the workers to stop work on the system pending a review of the situation.

Through discussions with RG&E operations and maintenance personnel, the inspector determined the procedural noncompliance occurred when two separate tasks-movement of fuel in the spent fuel pool, and alignment of the spent fuel pool ventilation system, were not properly sequenced. Specifically, operations department personnel signed off steps 3.4 and 3.5 of procedure M-7.9, which indicated plant conditions were acceptable to allow maintenance to be performed on the ventilation system, when in fact they subsequently were not. These errors were not identified by maintenance personnel when they received procedure M-7.9 from the operations department.

Analysis. The performance deficiency in this event is a failure of several people in multiple departments to properly sequence work activities, and follow steps in the applicable maintenance procedure. This finding associated with the Barrier Integrity Cornerstone, is greater than minor because it is similar to example 2.h of NRC manual Chapter 0612, ”Power Reactor Inspection Reports” where there were multiple examples of personnel failing to follow procedures. In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted an SDP Phase 1 screening and determined that the finding was of very low safety significance (Green). The safety significance was not greater than Green since although irradiated assemblies were being moved in the spent fuel pool, they had decayed greater than 60 days which significantly diminished the amount of radioactive material that could be released in the event of an assembly drop incident.

Enforcement. Plant TS 5.4. “Plant Procedures” states, in part, that “Written procedures shall be established, implemented, and maintained covering . . . the applicable procedures recommended in Regulatory Guide 1.33, Revision 2 Appendix A, February 1978.” Regulatory Guide 1.33, “Quality Assurance Program Requirements” identifies that procedures regarding the startup, shutdown, and changing modes of operation of the Auxiliary Building Heating and Ventilation System should be implemented. Contrary to the above, not all steps of procedure M-7.9 were implemented by Ginna personnel when the spent fuel pool ventilation system was being realigned. Because this procedure noncompliance was of very low safety significance, and was entered into the Ginna corrective action program under AR 2003-1699, “Procedures M-7.9 and AF-8.4 Initial Conditions Conflict,” this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000244/2003006-04 Did Not Follow Procedures for Maintenance on Spent Fuel Pool System Charcoal Filtration System.**

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

One month prior to plant shutdown, the inspectors reviewed the outage plan to verify RG&E had identified risk significant activities, and developed contingency plans to cope with those events. As part of the preparatory work for the outage, the inspectors reviewed new fuel receipt procedures, and observed unpackaging of new fuel including receipt inspection and transfer to the spent fuel pool.

On September 15, 2003, the inspectors observed control room and auxiliary operators shutdown the plant and perform an overspeed test of the main turbine. Once the plant was shutdown, the inspectors entered the containment and verified RG&E personnel had identified deficient conditions including valve packing and flange leakage, and potential RHR sump clogging items.

Prior to movement of fuel from the reactor vessel, the inspectors verified that control room and refueling bridge operators were monitoring the refuel cavity inflatable seal to ensure it was performing acceptably, and that contingency plans were in place if the air supply to the seal was lost. The inspectors also verified that containment integrity had been established in accordance with RG&E procedures, and potential leak paths were corrected.

The inspectors observed portions of fuel shuffle operations, and verified operators were following procedures. These activities accounted for partial completion of this inspection procedure.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed the performance and/or reviewed test data for the following six samples to verify that the tests demonstrate the associated system's functional capability and operational readiness:

- PT-2.1Q, "Safety Injection System Quarterly Test" performed on July 14, 2003.
- PT 3.1Q, "Containment Spray Pump Quarterly Test" performed on July 15, 2003.
- PT-16Q-B, "Auxiliary Feedwater Pump B - Quarterly" performed on July 7, 8, and 10, 2003.
- PT-2.2Q, "RHR Pump Operability" performed on July 28, 2003.
- PT-37.3, "Control Room Vent Mass Air Flow Check" performed on August 7, 2003.
- PT-38.1, "Control Room Filter Inspection" performed on August 6, 2003.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The following temporary modification (TM) was reviewed and visually inspected by the inspectors to verify that the TM was installed in conformance with the instructions contained in procedure IP-DES-3, “Temporary Modifications”: One sample was completed.

- 2003-0015, “Control Room Toilet Exhaust Damper Outlet Flex Joint, SC151 Repair”

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP3 Emergency Response Organization (ERO) Augmentation Testing (71114.03)

a. Inspection Scope

The NRC documented an unresolved item (URI) in Inspection Report 50-244/02-09 (URI 50-244/02-09-02) concerning RG&E’s Nuclear Emergency Response Plan (NERP) staffing commitments that were inconsistent with those prescribed in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." This constituted a potential failure to meet planning standard 10 CFR 50.47(b)(2) which states, in part, that adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, and timely augmentation of response capabilities is available.

The issue was referred to NRC headquarters specifically the office of Nuclear Reactor Regulation (NRR) via Task Interface Agreement 2002-02 on June 19, 2002, for review and resolution. NRR staff review determined that the on-shift and augmentation staffing levels described in the Ginna NERP, Revision 20, was not acceptable in implementing the requirements of 10 CFR 50.47(b)(2). In response, RG&E submitted proposed NERP enhancements by letter dated May 23, 2003. The proposed enhancements involved NERP changes to clarify minimum on-shift staffing levels, compensate for the lack of 30-minute augmentation staff, and revise the number of one-hour emergency responders. One sample was completed.

The NRR staff reviewed the proposed enhancements to the Ginna NERP and supporting documentation. They concluded that RG&E’s proposed Ginna NERP enhancements meet the standards of 10 CFR 50.47(b) and the requirements of 10 CFR 50, Appendix E, and are therefore acceptable. The staff documented their conclusion and its basis in an NRC letter dated July 24, 2003, which contained the Safety Evaluation by enclosure.

Therefore, URI 05000244/2002009-02 is closed.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector conducted an in-office and on-site review of RG&E-submitted changes for the emergency plan-related documents to determine if the changes decreased the effectiveness of the plan. A thorough review was conducted of documents related to the risk significant planning standards (RSPS), such as classifications, notifications, and protective action recommendations. A general review was conducted for non-RSPS documents. These changes were reviewed against 10 CFR 50.54(q) to ensure that the changes do not decrease the effectiveness of the plan, and that the changes as made continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E. These changes are subject to future inspections to ensure that the impact of the changes continues to meet NRC regulations. The submitted and reviewed documents are listed as attached. One sample was completed.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

During the period July 7 - 10, 2003, the inspector conducted the following activities to verify that RG&E had properly implemented physical and administrative controls for access to locked high radiation areas and other radiologically controlled areas, and that workers adhered to these controls when working in these areas. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, Technical Specifications, and RG&E's procedures.

- Independent radiation surveys were performed in the auxiliary building and intermediate building (Hot Side) to confirm the accuracy of posted survey results, and assess the adequacy of radiation work permits (RWP), associated controls, and area postings. The Radiation Protection Manager accompanied the inspector during the plant walkdown, and discussed with the inspector the adequacy of radiological controls established for these areas.

- Keys to technical specification locked high radiation areas (TSLHRA) were inventoried, and accessible areas were verified to be properly secured and posted during plant tours. Also reviewed were controls for highly activated or contaminated non-fuel materials stored in the spent fuel pool. To identify changes that could substantially reduce the effectiveness and level of worker protection, the inspector reviewed the high radiation area access control program.

- The inspector reviewed the RWPs and associated radiation survey maps for selected jobs performed during the inspection period; observed aspects of these work activities; and interviewed workers on their knowledge of the relevant RWP, electronic dosimetry setpoints, and job site radiological conditions. The inspector verified that radiological controls such as required surveys, technician job coverage, and contamination controls were implemented. The review included assessment of possible radiation dose gradients and the proper positioning of dosimetry. The inspector attended the pre-job RWP briefing for the spent fuel pool filter change-out. The observed work activities included:

    - Replacement of Spent Fuel Pool Filters (RWP 03-1024
    - Replacement of Limit Switches on AOV-966C (RWP 03-0001)
    - Auxiliary Operator Performing Routine Rounds (RWP 03-1001)

- The inspector reviewed pertinent information regarding cumulative exposure history for 2002, current exposure trends, and ongoing activities in order to assess RG&E's effectiveness in establishing exposure goals, controlling access to the work area, and minimizing worker dose.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

During the period September 22 -25, 2003, the inspector conducted the following activities to verify that RG&E properly implemented operational, engineering, and administrative controls to maintain personnel exposure as low as is reasonably achievable (ALARA) for tasks conducted during the refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and RG&E's procedures.

Radiological Work Planning

- The inspector reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing activities to assess current performance and exposure challenges. The inspector determined the plant's three-year rolling collective average exposure.

- The inspector reviewed the refueling outage work scheduled during the inspection period and the associated work activity exposure estimates. Scheduled work reviewed included: reactor vessel head replacement activities, refueling cavity concrete wall repair, removal/ cutting of control rod drive mechanisms from the old reactor vessel head, and fuel transfers.

- The inspector reviewed procedures associated with maintaining worker dose ALARA and with estimating and tracking work activity specific exposures.

- The inspector reviewed the outage ALARA Review summary list, which detailed the worker estimated and actual exposures, through September 25, 2003, for jobs performed during the refueling outage.

- The inspector evaluated the exposure mitigation requirements, specified in ALARA Reviews (AR), and compared actual worker cumulative exposure to estimated dose for tasks associated with these work activities. Jobs reviewed included: Valve Maintenance (AR 03-0062), Reactor Vessel Inspection of Lower Penetrations (AR 03-0202), and Reactor Head Replacement Work (AR 03-0701).

- The inspector evaluated the departmental interfaces between radiation protection, operations, maintenance crafts, engineering, and in-service inspection groups, to identify missing ALARA program elements and interface problems. The evaluation was accomplished by interviewing the ALARA Coordinator, reviewing ALARA Committee Meeting minutes, and attending pre-job briefings for jobs in progress.

- The inspector compared the person-hour estimates provided by maintenance planning and other work groups with actual work activity time requirements and evaluated the accuracy of these time estimates. Specific work activities evaluated included: Valve maintenance (AR 03-0062), Routine Maintenance inside Containment (AR 03-0077), Filter Changes (AR 03-0104), Reactor Vessel inspection of Lower Penetrations (AR 03-0202), and Reactor Head Replacement Work (AR 03-0701).

- The inspector determined if work activity planning included the use of temporary shielding, system flushes, and operational considerations; i.e., adjusting steam generator water levels, to further minimize worker exposure. The inspector reviewed Temporary Shielding Requests (Nos. 3-10, 3-23, 3-24, 3-32) and survey results for flushes of the residual heat removal system and the auxiliary building sump tank piping.
- The inspector reviewed the ALARA In-Progress Reviews for the Reactor Head Replacement and the under vessel inspections in "A" sump to determine if revised dose projections were properly justified. Additionally, the inspector

evaluated the Post-job ALARA Review for replacing the source range detector (N-32) to determine if worker problem areas were being identified and that lessons learned from the activity were being addressed.

Verification of Dose Estimates and Exposure Tracking Systems

- The inspector reviewed the assumptions and basis for the current annual collective exposure estimate and the refueling outage dose projection.

- The inspector reviewed RG&E's method for adjusting exposure estimates, and re-planning work, when emergent work was encountered.

- The inspector reviewed RG&E's exposure tracking system to determine whether the level of dose tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support the control of collective exposures. Included in this review were the radiation work permits (RWP) for inspecting fuel handling equipment (RWP 03-1046), inspecting/ repairing the B-sump screens (RWP 03-1065), and performing leak repair on refueling cavity walls (RWP 03-1066).

Job Site Inspection and ALARA Control

- The inspector observed maintenance and engineering activities being performed on CRDM cutting and cavity wall leak repairs to verify that radiological controls, such as required surveys, job coverage, and contamination controls were implemented; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological conditions.

- The inspector reviewed the exposures of individuals in selected work groups, including operations, mechanical maintenance, radiation protection, and engineering to determine if supervisory efforts were being made to equalize doses among the workers.

Source Term Reduction and Control

- The inspector reviewed the current status and historical trends of the plant's source terms. Through interviews with the Chemistry Manager and the ALARA Coordinator, the inspector evaluated RG&E's source term control strategy. Specific strategies being employed by RG&E include post shutdown peroxide flushes of reactor coolant piping and pre-startup flushes of the residual heat removal system.

Radiation Worker Performance

- The inspector observed radiation worker and radiation protection technician performance during the disassembly of the old reactor vessel head and the seal injection filter replacement, and determined whether the individuals were aware of radiological conditions, RWP requirements, and EPD set points; and that the skill level was sufficient with respect to the radiological hazards and the work involved.

- The inspector attended the pre-job briefings for exposure-significant tasks performed during the inspection period to determine the adequacy and accuracy of information provided to workers. Pre-job briefings attended included refueling cavity wall leak repairs and CRDM removal/ cutting from the old reactor head.

- The inspector reviewed problem reports related to radworker or radiation protection technician errors to determine if an observable pattern traceable to a similar cause was evident.

Declared Pregnant Workers

- The inspector determined if there have been any declared pregnant workers (DPW) during the current assessment period. The exposure results and monitoring controls for two (2) DPWs were reviewed.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP2 Access Control (71130.02)

a. Inspection Scope

The following activities were conducted during the inspection period to verify that RG&E had effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area as measured against 10 CFR 73.55(d) and the Physical Security Plan and Procedures:

- On August 5, 2003, safeguards log entries and event reports for the previous twelve months associated with the Access Control Program were reviewed. A review was conducted on August 6, 2003 of the testing and maintenance procedures used to conduct periodic performance testing of all search equipment to determine if the testing program was sufficiently challenging, and implemented in accordance with the Physical Security Plan and associated procedures.

- Site access control activities were observed, including personnel and package processing through the search equipment during peak egress periods on August 5 and August 6, 2003. On August 6, 2003, observation of vehicle search activities was also conducted. On August 6, 2003, testing of all access control equipment; including metal detectors, explosive material detectors, and X-ray examination equipment was observed.

- On August 7, 2003, a review of the annual security audit and several self-assessment documents was conducted, to verify that any issues associated with the access control and search programs were entered into the corrective action program as appropriate, and that these issues were effectively resolved.

b. Findings

No findings of significance were identified.

3PP3 Response to Contingency Events (71130.03)

a. Inspection Scope

The following activities were conducted to determine the effectiveness of Ginna's response to contingency events, as measured against the requirements of 10 CFR 73.55 and the Ginna Safeguards Contingency Plan:

- On August 7, 2003, a review of documentation associated with the Ginna force-on-force exercise program was conducted. The review included documentation of training exercises conducted since the first quarter of 2002, when the exercises were resumed post September 11, 2001.

- On August 6, 2003, performance testing of the Ginna intrusion detection and alarm assessment systems was conducted. This testing was accomplished by one inspector who toured the plant perimeter and selected, and subsequently observed performance tests, of areas of potential vulnerability in the intrusion detection system. Concurrently, a second inspector observed both the audible alarms and the alarm assessment capabilities from the central alarm station. During the walkdown of the intrusion detection system, all 29 zones were performance tested, by a combination of 29 walk, 8 run and 29 crawl tests.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Radiation Safety/Occupational Radiation Safety Cornerstone

The inspector reviewed implementation of RG&E's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspector reviewed recent Action Reports, and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned personnel exposures against the criteria specified in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, to verify that all occurrences that met the NEI criteria were identified and reported as Performance Indicators.

Safeguards/ Protection Cornerstone

On August 7, 2003, a review was conducted of RG&E 's programs for gathering, processing, evaluating, and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators (PIs) to verify these PIs had been properly reported as specified in NEI 99-02. The review included RG&E's tracking and trending reports, personnel interviews and security event reports for the PI data collected from the 2nd quarter of 2002 through July 2003.

Reactor Safety/ Emergency Preparedness Cornerstone

The inspector reviewed RG&E's process for identifying the data that is utilized to determine the values for the three Emergency Preparedness performance indicators (PI) which are: 1) Drill and Exercise Performance, 2) Emergency Response Organization (ERO) Participation, and 3) Alert Notification System (ANS) Reliability. The review assessed data submitted to the NRC from the second quarter of 2002 (since the last EP PI verification inspection) up to, and including, the second quarter of 2003. Classification, notification, and protective action opportunities were reviewed from licensed operator simulator sessions and site ERO drills and exercises. Attendance records for drill and exercise participation were reviewed for verification purposes. Test results of the ANS testing were reviewed for accuracy and completeness. The inspector reviewed this data using the criteria of NEI 99-02.

Reactor Safety/ Mitigating Systems

The inspector sampled RG&E submittals for the performance indicators (PIs) listed below. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02 were used.

Reactor Safety Cornerstone

- Safety System Functional Failures, October 2002 through June 2003
- Safety System Unavailability - Emergency AC Power, first quarter 2002, third quarter 2002, forth quarter 2002, first quarter 2003, and second quarter 2003

To perform this review, the inspectors reviewed main control room records, corrective action program records and work orders, and compared them to the monthly PI data reports and forms.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Review of Reactor Vessel Closure Head Replacement Documentation

a. Inspection Scope

The inspectors reviewed corrective action documents associated with RVCH 10 CFR 50.59 and plant modification issues to ensure that RG&E appropriately identified, evaluated, and initiated actions to correct problems associated with these activities. The inspectors also reviewed a Quality Assurance (QA) audit and RG&E self-assessments related to the RVCH 10 CFR 50.59 evaluations and plant modifications.

b. Findings

No findings of significance were identified.

2. Review of Spent Fuel Pool (SFP) Water Leak Action Reports

a. Inspection Scope

The inspector reviewed the following documents to ensure that the corrective actions for the associated plant issue were appropriate. This issue was selected for follow-up review due to its potential safety significance.

- Safety Evaluation SEV-1123, “Spent Fuel Pool Leakage Release Pathway Assessment (April 1999)”
- Action Report (AR) 2001-2100, “Water in RHR PIT Coming From Ceiling”
- AR 2002-2456, “AR 2001-2100 Closed Without the Maintenance Rule Review”
- Work Order No. 20203250, “Leak Repair in RHR PIT”
- Toured RHR PIT area

- Technical Staff Request (TSR) 2003-0046, “RHR Sub-basement In-leakage and Restoration”
- Trending Evaluation of the SFP Water Leak Rate
- Review of two new onsite H-3 Monitoring Wells
- Trending Evaluation of the onsite H-3 Monitoring Wells (6/2001- 8/2003 and associated procedures.)

b. Findings

No findings of significance were identified. RG&E made the following determinations based on the engineering evaluations of SFP systems, the effectiveness of leak repair efforts, tritium measurements of onsite wells, and assessment of site-specific hydrology.

- The average SFP water leak rate is about 1 gallon per day
- Tritium measurements of onsite wells were lower limit of detection (LLD)
- Ground water flow rate is extremely slow
- There is no evidence that tritium has migrated beyond the radiological restricted area

3. ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed fourteen Action Reports and ten Quality Assurance Surveillance Observations, relating to controlling worker exposures, to evaluate the licensee’s threshold for identifying, evaluating, and resolving problems relating to occupational radiation safety. The review included a check of possible repetitive issues such as radiation worker or radiation protection technician errors.

This review was conducted against the criteria contained in 10 CFR 20, Technical Specifications, and RG&E’s ALARA-related procedures.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

August 14, 2003 Trip Notice of Enforcement Discretion

On August 15, 2003, RG&E applied for a Notice of Enforcement Discretion (NOED) that would allow Ginna to change modes without an operable “B” MDAFW pump. The circumstances that led up the “B” MDAFW pump being rendered inoperable are discussed in section R14 of this report. Enforcement discretion was granted verbally by the NRC on August 15, 2003, and by written correspondence in an August 20, 2003 letter to RG&E. To obtain the enforcement discretion, RG&E committed to perform several actions, including taking steps which ensured operability of components on the opposite train of safety-related equipment. On a sampling basis, the inspectors verified RG&E completed the actions that they committed to perform in the August 15, 2003 letter. Following issuance of the August 20, 2003, letter, Unresolved Item (URI) 05000244/2003006-03 was opened to track NRC followup of the issues that led to the need for the NOED. In section R14 of this report, the reviewed the circumstances that led to failure of the “B” MDAFW pump, as such Unresolved Item (URI) 05000244/2003003006-03 is closed.

4OA4 Cross-Cutting Aspects of Findings

Section 1R14 discusses a human performance contributing cause of a finding. Inadequate placekeeping in the procedure by the operating crew resulted in the omission of the step in the procedure to shutdown the “B” motor driven auxiliary feedwater pump.

4OA5 Other Activities

1. Pre-Service Inspection and Testing (71120)

a. Inspection Scope

By September 8, Framatome and RG&E had completed the assembly of the new RVCH, CRDMs, and related components (see Section 1R02). During the week of September 8, the inspectors reviewed the extent of nondestructive examination (NDE) performed during fabrication and assembly of the head and its components. This review included the ASME Code acceptance NDE for construction and that done as pre-service inspection (PSI) to provide a comparative basis for evaluation of the CRDMs by NDE after it has been in service. The ASME NDE review included: samples of the ultrasonic tests (UTs) of the head forging, radiography tests (RTs) of the CRDM guide tube to CRDM adapter No. 85 welds, and visual and penetrant test (PT) results of seal welds. The PSI examination review included the UT from the CRDM guide tube (inner diameter) and eddy current test (ECT) of the weld surface area of CRDM guide tubes to the head internal surface. The inspectors observed a PT examination of CRDM to head welds No. 1 and 6 to confirm that no change in the as-welded surface condition had occurred between the time of final welding in Canada and September 10, 2003, at the Ginna plant.

Additionally, the inspectors reviewed portions of the data packages for the head forging, head to CRDM guide tube welding, and the assembly of the CRDMs to determine if the data documentation was appropriate, accurate, and that documentation problems were identified in the corrective action program for resolution.

b. Findings

No findings of significance were identified.

2. Review of Engineering Design, Modification, and Analyses

a. Inspection Scope

The inspectors reviewed the analyses, design calculations, and evaluations for head component drop, lay-down area, and safe load path for the RVCH movement and storage in containment. The inspectors reviewed the applicable documents (e.g., Turbine Building Floor Structural Analysis, Runway System) for moving the new head out of the turbine building and into the containment. This review also focused on the potential impact of load handling activities on the reactor core, spent fuel and the spent fuel pool cooling system, and other plant support systems. The inspectors observed that RG&E made no major structural modifications associated with the RVCH replacement activity and did not need any temporary modifications for the containment access.

b. Findings

No findings of significance were identified.

3. Lifting and Rigging of the New Reactor Vessel Closure Head

a. Inspection Scope

The inspectors reviewed the activities associated with rigging and lifting of the new RVCH. The review included: preparations and procedures for rigging and heavy lifting, required crane and lifting devices inspection, testing, required structural and equipment modifications, preparation of lay-down area, and training of rigging personnel. The inspectors verified that the capability of the lifting equipment, including fixtures and rigging, had been analyzed and evaluated though engineering calculations and analyses. In addition, the inspectors observed RG&E move the new RVCH from the temporary lay-down area to the containment.

b. Findings

No findings of significance were identified.

4. Control Rod Drive Mechanism Record Review

a. Inspection Scope

On April 1, 2002, RG&E issued purchase order 4500019441 to Framatome ANP, Inc. for the supply of 29 CRDM assemblies. The inspectors reviewed the manufacturing records for two CRDMs (Serial Nos. 2591 and 2593). This record review included the certified material test certificates attesting to the quality of the material and the various processes used in CRDM manufacture. The inspectors reviewed the records to verify that Framatome manufactured the CRDMs using a quality assurance program and in compliance with the requirement of 10 CFR 50, Appendix B.

b. Findings

No findings of significance were identified.

5. Reactor Vessel Closure Head Pipe Support Installation Review

a. Inspection Scope

The inspectors reviewed the records documenting the installation of pipe supports to the RVCH radiation shield to ensure that RG&E followed the applicable procedures and performed and adequately documented necessary quality control inspections. The pipe support weld reviews included: the design drawing and location of the support, verification of materials used, pipe gap distances, the weld procedure specification (WPS) used, the joint preparation inspection, the joint cleanliness, the interpass temperature, the final visual inspection, and the liquid penetrant examination of the weld surface.

b. Findings

No findings of significance were identified.

6. TI 2515/152 - Reactor Pressure Vessel Lower Head Penetration Nozzles

a. Inspection Scope

The inspectors reviewed RG&E's activities in response to Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," as required by TI 2515/152 for pressurized water reactors. This included interviews with analyst personnel as well as a review of qualification records and plant inspection procedures. Additionally, the inspectors independently reviewed the results of the visual examination, both directly at the reactor vessel lower head and by videotape.

In accordance with TI 2515/152, the inspectors verified that deficiencies and discrepancies associated with the reactor coolant system structures, such as boric acid deposits, were identified and assured that they were placed in RG&E's corrective action process. The inspectors reviewed RG&E's assessment of boric acid residue found on the lower head, which was attributed to reactor cavity seal leakage. This included a review of RG&E's chemical analysis of the deposits.

b. Findings

No findings of significance were identified.

The following input addresses the specific reporting requirements of TI 2515/152:

1. The examination was performed by qualified and knowledgeable personnel. A review of personnel qualification records indicated that the personnel performing the visual inspection were VT-1 and VT-3 qualified.

2. The visual examination was conducted in accordance with approved and adequate procedures.

3. The examination was adequate to identify, disposition, and resolve deficiencies.

4. The examination performed was capable of identifying the pressure boundary leakage as described in Bulletin 2003-02.

5. The general condition of the reactor vessel (RV) head was clean metal with a layer of gray Carbo-Zinc paint covering the bottom head and the upper portion of some of the nozzles. There was come localized boric acid staining that the licensee attributed to past reactor cavity seal ring leakage. Faint streams of boric acid residue were visible coming down from above the lower head and around several nozzles. There was little or no debris or dirt on the lower head. RG&E had completely removed the insulation package from the lower head and erected scaffolding to provide access. This arrangement allowed for 360° visual coverage around the circumference of all penetration nozzles. There were no significant viewing obstructions.

6. Small boron deposits, as described in Bulletin 2003-02, were able to be identified and characterized. None were found during this visual inspection.

7. No material deficiencies associated with concerns in Bulletin 2003-02 were found.

8. Site ALARA controls were effective at minimizing unnecessary or unexpected dose to personnel. Dose rate considerations should not preclude or impede future examinations. Past and future reactor cavity seal ring leakage was the only identified item observed during the inspection that could potentially challenge effective examinations in the future. However, it was concluded that the seal leakage to date does not mask leakage from nozzle penetrations.

9. The inspectors verified that RG&E conducted follow-on examinations for indications of boric acid leaks from pressure-retaining components above the reactor vessel lower head.

7. Radiological Aspects of Reactor Vessel Head Replacement

a. Inspection Scope

The inspector evaluated various activities to verify that adequate radiological safety was maintained during RV head removal and replacement activities. The inspector reviewed planning activities for the reactor vessel upper head replacement scheduled to be performed during the Fall 2003 refueling outage. The following matters were reviewed:

- The inspector attended a pre-job briefing for removal and cutting of control rod drive mechanisms from the old reactor head, to evaluate the adequacy of radiological controls applied to this work activity.

- The inspector reviewed the characterization plan and radiological survey data obtained from the old reactor vessel head and associated components that would be used for characterizing the radio-isotopic content of the old head in preparation for its disposal.

- The inspector observed workers installing components on the new head.

- The inspector observed workers performing their assigned tasks relative to disassembling the old reactor head.

- The inspector reviewed the contractor's ALARA Plan for replacing the reactor vessel closure head.

- Planned occupational exposure goals for various phases of the reactor head replacement

- Project schedule and ALARA planning and contamination controls for reactor head transfer and storage

- Station ALARA Committee Meeting minutes that relate to the reactor vessel head replacement project

This review was conducted against the criteria contained in 10 CFR 19, 10 CFR 20, site Technical Specifications, and applicable site procedures.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Periodically during the course of this inspection, the inspectors met with Ginna representatives to discuss certain aspects of the inspection. For example, on August 7, 2003, the purpose and scope of the physical security inspection were reviewed, and the preliminary findings were presented. On August 21, 2003, via teleconference, the final results were presented to RG&E management, who agreed with the facts presented at the exit.

On October 16, 2003, the resident inspectors summarized the contents of this inspection report to Mr. Widay, and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-identified Violations

The following violations of very low safety significance (Green) were identified by RG&E and were violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV:

- TS 5.4.1 requires that procedures for radiation safety be implemented. Contrary to this requirement, on June 24, 2003, radiation safety procedure RP-TLD-142-10-OPS, Rev. 2, was not implemented. Specifically, the operator of a Panoramic Irradiator did not verify if personnel were working on the roof, above the irradiator, before exposing the source. When the source was exposed, a worker was on the roof and was not wearing dosimetry. The worker had the potential of receiving a radiation dose for which radiation monitoring was required. However, the worker and his escort immediately left the area when they saw a local warning beacon indicating the source had been exposed. This finding is greater than minor in that it is associated with the Occupational Radiation Safety Cornerstone and did affect the cornerstone objective for ensuring worker protection from radiation. The finding is of very low safety significance because it was not an ALARA issue, did not involve a High Radiation Area, did not result in a substantial potential for a personnel over-exposure, and did not compromise the ability to assess dose. This finding is in RG&E's corrective action program as AR 2003-1379.

- 10 CFR 50 Appendix B Criterion V, "Instructions Procedures and Drawings" states, in part, that "Activities affecting quality shall be prescribed by documented instructions procedures or drawings . . . appropriate to the circumstances . . . Contrary to this requirement, RG&E did not provide maintenance personnel with appropriate instructions for rebuilding the lube oil circulating for the "A" motor-driven auxiliary feedwater pump. As a result, the reliability of the pump was reduced and might have been out of service for greater than its TS allowed outage time. This finding is in RG&E's corrective action program as AR 2003-2006.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

RG&E personnel

P. Bamford	Operations Manager
M. Flaherty	Nuclear Safety & Licensing Manager
B. Flynn	Primary Systems and Reactor Engineering Manager
J. Hotchkiss	Mechanical Maintenance Manager
G. Jones	Radio-chemist, Primary Systems
T. Laursen	Corporate Emergency Preparedness Manager
R. Marchionda	Nuclear Assessment Department Manager
B. Mecredy	Vice President Nuclear Operations
F. Mis	Manager, Chemistry
P. Polfleit	Corporate Emergency Preparedness Planner
R. Popp	Production Superintendent
J. Smith	Maintenance Superintendent
W.Thomson	Manager, Radiation Protection
T. White	Balance of Plant Systems Engineering Manager
J. Widay	VP, Plant Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000244/2003006-01	FIN	No Procedure Guidance for Loss of Relay Room Cooling (Section 1R13)
05000244/2003006-02	NCV	Operators Did Not Shutdown “B” MDAFW pump per ES-0.1 (Section 1R14
05000244/2003003-03	FIN	Vendor Manual Control Program was Inadequate Section 1R19)
05000244/2003006-04	NCV	Did Not Follow Procedures for Maintenance on Spent Fuel Pool System Charcoal Filtration System. (Section 1R19)

Opened and Closed

05000244/2003006-06	URI	Notice of Enforcement Discretion Issued that allowed a mode change with an inoperable “B” MDAFW pump. (Section 4OA3)

Closed

05000244/2002009-02	URI	Licensee E-Plan staffing commitments were inconsistent with those prescribed in NUREG-0654, which was a potential finding for failure to meet planning standard 10 CFR 50.47(b)(2). (Section 1EP3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluation of Changes, Tests, or Experiments

Documents Reviewed

Reactor Vessel Closure Head Replacement (PCR-0042), Rev. 0
R.E. Ginna Station, Core Exit Thermocouple Nozzle Assembly (EVAL-03-42)
R.E. Ginna Reactor Vessel Head Area Upgrade Package (EVAL-03-23)
Reactor Vessel Closure Head Replacement (5059SCRN 2003-0271)
Ginna Station Part Length CRDM Drive Rod Elimination (EVAL-03-81)
Ginna Nuclear Power Plant CRDM Pressure Housing Assembly Appurtenances ASME III Class 1 Design Report (6 CS 1075)

Section 1R05: Fire Protection

Action Reports

2003-1818, Questionable Seismic Mounting of Appendix R Light in Control Room

Section 1R07: Heat Sink Performance

Action Report

2003-2050 Steady Stream of Water From "C" Recirc Fan Cooler

Analysis

EWR 5275 Containment Recirculating Fan Cooler Replacement

Procedures

S-23.6	"Containment Recirculation Fan Cooler Motor Flush"
CMP-10-07	"Marlo Model 12Q Cooling Coil Inspecton and Maintenance for ACA01A, AC01E, and ACA01F"
RSSP-2.4	"CNMT Recirculation Fan Service Water Leak Check"

Work Orders

20301751	Water Dripping From Bottom of Cooler
20203774	Flush ACA07 and ACA10
20101523	Perform UT on Copper Tubing on ACA10
19703105	Wash Dirt and Fuzz From All Four CRFC Units
19804036	1/4" Pipe Plugs in Coolers
19702840	Clean Coolers

Section 1R08: Inservice Inspection Activities

Documents

NDE-UT-208, Manual UT Examination of Austentic Pressure Piping Welds (PDI)
NDE-PT-106, Liquid Penetrant Exams
MT Summary Number I201051
RT Summary Number I200220
RT Summary Number I200160
Report Number BOP-RT-03-061
Report Number BOP-RT-03-055
Report Number BOP-RT-03-062
Report Number 03GRT054M
B&W Drawing 33013-2835, Revision 0, Containment Refueling Cavity and Spent Fuel Pit Volumes (Cross-Sectional View)
Post-LOCA Containment Hydrogen Generation Evaluation for the R.E. Ginna Nuclear Power Station
AR 2003-2312, Error in Containment Post-LOCA Hydrogen Calculation

Section 1R12: Maintenance Rule Implementation

Action Reports

2002-0200, 3B Low Pressure Feedwater Heater Relief Valve Stuck Open
2002-0205, Relief Valve Stuck Open After Reactor Trip
2002-0201, 1B/2B Low Pressure Feedwater Heater Relief Stuck Open
2002-0202, 4B Low Pressure Feedwater Heater Relief Stuck Open
2002-0203, 3A Low Pressure Feedwater Heater Relief Stuck Open
2001-1324, B Main Feedwater Pump Suction Relief Leaking By
2003-1805, Relief Stuck Open, B Main Feedwater Pump
2003-1806, Relief Stuck Open, 3B Heater
2003-0043, Total RCP Seal Leakoff Flow >5.74 GPM
2003-0453, Excessive Charging Flow Demand With 40 GPM Orifice In Service
2003-0493, RV-284 Premature Lift Setpoint
2003-0477, Excessive Plunger Leakage B Charging Pump
2003-0534, Minimum Charging For B Charging Pump Does Not Meet PT-31 Requirements
2003-0735, C Charging Pump Failed to Meet PT-31 Requirements
2003-0705, A Charging Pump Failed to Meet PT-31 Requirements
2002-0016, Abnormal Noise from C Charging Pump Belt Housing
2002-0161, B Charging Pump Running Slower Than A Charging Pump

2002-0296, C Charging Pump Relief Valve RV-283 Lifting
2002-0511, AOV-392A Failed To Close
2002-0695, CU-304B Failed Leak Test
2002-1014, Cracked Weld On C Charging Pump Discharge Line
2002-1398, Excessive Leakage Noted ON A Charging Pump
2002-1557, Plunger Assembly Throat Bushing Cracked
2002-2383, B Charging Pump Failed Minimum Flow Output of PT-31

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Action Reports

2003-1989, Yard Loop Main Leak at Access Road

Work Orders

WO 2030176 3 L-7 Alarmed at 0339; MCB Indication for Bus 16 Normal at 480V, Targets Tripped for 27/16 and 27D/16

Section 1R14: Personnel Performance During Non-routine Plant Evolutions

Action Reports

2003-1745, No Procedure Guidance for Loss of Relay Room Cooling
2003-1804, "B" AFW Pump Overheating
2003-1812, Reactor Trip Following Grid Problems
2003-1815, Bus 17 Undervoltage Power Supply
2003-1821, "B" AFW Pump Damage Human Performance Issues
2003-1830, Rod Control Step Counters Not Resetting
2003-1833, MCB Alarm H-9, Auxiliary Feed Pump Cooling Water
2003-1837, Control Unit Air Handling Unit Tripped
2003-1840, Urgent Failure Alarm
2003-1841 Tripped Turbine on High Back Pressure

Section 1R15: Operability Evaluations

Action Reports

2003-0309, AOV 966C Exceeds Administrative Leakage Limit
2003-1489, Service Water Expansion Joint Tie-rods
2003-1550, SW Leak on V-4619
2003-1720, Control Room Roof Leakage
2003-2024, Pump Minimum Flow Output Not Obtained B Charging Pump
2003-1933, Spent Fuel Pool Check Valve Failure

Section 1R17: Permanent Plant Modifications

Documents Reviewed

Reactor Vessel Closure Head Replacement (PCR-0042), Rev. 0

R.E. Ginna Station, Core Exit Thermocouple Nozzle Assembly (EVAL-03-42)
R.E. Ginna Reactor Vessel Head Area Upgrade Package (EVAL-03-23)
Reactor Vessel Closure Head Replacement (5059SCRN 2003-0271)
Ginna Station Part Length CRDM Drive Rod Elimination (EVAL-03-81)
Ginna Nuclear Power Plant CRDM Pressure Housing Assembly Appurtenances ASME III Class 1 Design Report (6 CS 1075)
Framatome ANP Document Submittal - Welder Certifications (FANP-03-2446), dated August 13, 2003

Procedure Qualification Record 035N005 (for qualifying nameplate to latch housing welding)

Procedure Qualification Record 03SN001 (for qualifying welding of cap-to-rod travel housing)
Weld Procedure Qualification 03SN006 (for welding nameplate to latch housing)
Weld Procedure Qualification 03SN002 (for welding cap-to-rod travel housing)
Weld Procedure Qualification 03SN004 (for welding hatch housing to rod travel housing)
Weld Procedure Specification 76439F1, Rev. 1 (for welding cap-to-rod travel housing)
Weld Procedure Specification 764382F1, Rev. B (for welding nameplate to latch housing)
Pipe Support Inspection Record (for welded RVCH radiation shield pipe support Nos. RVLIS HS-1, RVLIS VS-1, RVH HS-1, PS-276-3-V1, PS-276-4-V2, and RVH VS-1)

Drawings

CRDM Ginna Outline for Specification (6MN1191)
Head Materials Drawing (083NA015), Rev. 04, 12 pages
RPV Closure Head Ordering (B&W Canada Drawing 083NE100), Rev. 02
The Arrangement of Reactor Vessel Longitudinal Sections (B&W Drawing 117802E), Rev. 7
Reactor Vessel Head Vent and RVLIS Isometric and Details Drawing (33013-2864), Sheet 1
Support No. RVLIS HS-1 RVLIS Support to Radiation Shield Drawing (10904-0693), Rev. 0
Support No. RVLIS VS-1 RVLIS Support to Radiation Shield Drawing (10904-0696), Rev. 0
Support No. RVH HS-1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0692), Rev. 0
Support No. PS-276-3-V1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0694), Rev. 0
Support No. PS-276-4-V2 Reactor Head Vent Support to Radiation Shield Drawing (10904-0695), Rev. 0
Support No. RVH VS-1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0697), Rev. 0

Plant Change Record
2000-0048, Smoke Detection Upgrades

Procedures

Reactor Head Hydrostatic Test Shop Instruction (259103), Rev. 1
Nondestructive Examination Control (A-903), Rev. 13
Control of Welding (A-901), Rev. 11
Radiographic Technique Sheet for Weld 85 (259123), Rev. 1
Framatome ANP Procedure CRDM Torquing and Welding Reactor Vessel Head Replacement (No. 7 MN 10924), Rev. A
Framatome ANP Nondestructive Examination Procedure Visible Solvent Removable Liquid Penetrant Examination Procedure (54-ISI-240-41), Rev. February 10, 2003
Framatome ANP Welding Components on Reactor Vessel Head Adapters Using Automatic Orbital GTAW Process Welding Machine Type ESAB ProTig 315 (6 MN 11911), Rev. B
Framatome ANP Engineering Verification of Reactor Vessel Closure Head Before and After Seal Welds Machine Type ESAB ProTig 315 (6 MN 1250), Rev. J

Other

ASME Code Form N-2 for the replacement RPV head forging, dated July 17, 2002
ASME Code Form N-2 for the replacement RPV head assembly, dated August 7, 2003
NDE Checklist for Ginna RVCH SN O83N-01, BWC-CONT-083N
RT Report No. 1 for SN 5210497, ref. hole No. 1, weld 85, dated November 14, 2002
RT Report No. 2 for SN 5211403-1, ref. hole Nos. 6, 12, 15, and 30, weld 85, dated February 18, 2003
RG&E Trip Report on the hydrostatic pressure test for the replacement RVCH, dated July 18, 2003
Framatome ANP Weld Control Record, Ginna Replacement RVCH Assembly (Outside Containment) (Process Traveler No. 50-5028903-01), dated August 14, 2003
Framatome ANP Procedure Test Specimen Welding Data Sheet (No. 7 MN 10924), dated August 20, 2003, August 22, 2003, and August 26, 2003
Certificates of Analysis for Argon Bottles (Cylinder Nos. 33-010268, 33-008148, 33-009272, and 33-007132)
Component - adapter welding data sheet and GTAW welding checklist, fit-up before welding, and liquid penetrant examination data for the following adapter/component welds: F12/2610, H4/2590, G13/2593, J3/2606, J13/2607, K4/2608, K8/2588, K12/2609, L5/2583, L11/2585, M6/2589, M8/2586, M10/2584, N7/2587, and N9/2582Analysis of Containment Floor at Elevation 274' 6" (Calc. No. 2060-C-7.1), Rev. 0
Overhead Door and Rotor (EWR No. 2192), Rev. 0
Turbine Building Structural Floor Framing to Support New Reactor Vessel Closure Head Replacement (DA-CE-2003-029), Rev. 0
Containment Building Crane Rigging for Old Head Lift (Calc. No. 2060-C3.3), Rev.1
Runway System (Calc. No. 2060-C4.1), Rev. 1
Containment Building New Head Rigging Test (Calc. No. 2060-C6.1), Rev. 2
Tie-down of Head to Transporter (Calc. No. 2060-C5.2), Rev. 0
Bigge Transporter Configuration, Pull Force and Wheel Loads (Calc. No. 2060-C5.1), Rev. 0
Ginna Turbine Building Runway System (Calc. No. 2060-C4.2), Rev. 1
Apex Plate Girder for HAUP & CRDM Support (Calc. No. 2060-C3.4), Rev.0
Rigging and Gantry/Crane for Loading or Offloading Bigge Transporter (Calc. No. 2060-C2.1), Rev. 1
Action Report (AR) No. 2003-2059
Incident Report 02NX0849 dated June 12, 2002

Receipt Inspection of Reactor Vessel Head Action Report No. 2003-1827
Framatome ANP Nonconformance Report No. 6028485
Framatome ANP Nonconformance Report No. 6028487
Framatome ANP Work Instruction WI-3 Condition Report No. 6028431

Section 1R19: Post Maintenance Testing

Action Reports

2003-1699, Procedures M-7.9 and AF 8.4 Initial Conditions Conflict
2003-2006, “A” AFW Pump Lube Oil Pressure Was Zero

Procedures

PT- 38.1, Visual Inspection of Charcoal Absorber Cell Assemblies
PT- 38.2, Visual Inspection of HEPA Filter Assemblies
IP- RDM-2, Vendor Technical Document Control and Change Requests
M-11.23, Worthington Double-Helical Rotary Pump Inspection and Maintenance
PT-16.3A, AFW Pump A Discharge MOV and Check Valve Test

Work Orders

20302211 Auxiliary Feedwater Pump “A”

Section 1R20: Refueling and Outage Activities

Action Reports

2003-1740, Potential Loss of Experienced Personnel For New Fuel Receipt

Procedures

A-3.3, ““Containment Integrity Program”

Section 1R22: Surveillance Testing

Action Reports

2003-1728, Frayed Corners on CRHVAC Flex Connectors

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Procedures

Ginna Station Nuclear Emergency Response Plan, Rev 22
EPIP 1-0, Ginna Station Event Evaluation and Classification, Rev 31
EPIP 1-1, Unusual Event, Rev 4
EPIP 1-5, Notifications, Rev 55
EPIP 1-6, Site Evacuation, Rev 16, 17
EPIP 1-8, Search and Rescue Operation, Rev 6

EPIP 1-9, Technical Support Center Action, Rev 24, 25
EPIP 1-10, Operational Support Center (OSC) Activation, Rev 13
EPIP 1-13, Local Radiation Emergency, Rev 5
EPIP 1-17, Planning for Adverse Weather, Rev 4
EPIP 2-1,Protective Action Recommendations, Rev 21
EPIP 2-4, Emergency Dose Projections - Manual Method, Rev 15
EPIP 2-9, Administration of Potassium Iodine (KI), Rev 8
EPIP 2-18, Control Room Dose Assessment, Rev 15
EPIP 3-1, Emergency Operations Facility (EOF) Activation and Operations, Rev 22, 23
EPIP 3-3, Immediate Entry, Rev 10
EPIP 4-1, Public Information Response to an Unusual Event, Rev 7
EPIP 4-3, Accidental Activation of Ginna Emergency Notification System Sirens, Rev 12, 13
EPIP 4-7, Public Information Organization Staffing, Rev 23
EPIP 5-1, Offsite Emergency Response Facilities and Equipment Periodic Inventory Checks and Tests, Rev 28
EPIP 5-2, Onsite Emergency Response Facilities and Equipment Periodic Inventory Checks and Tests, Rev 31
EPIP 5-5, Conduct of Drills and Exercises, Rev 15
EPIP 5-7, Emergency Organization, Rev 40
R.E. Ginna Emergency Action Levels Technical Basis, Rev 31
July 31, 2003, 10CFR50.54(q) for Revision 22 on the Nuclear Emergency Response Plan
July 24, 2003, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Nuclear Emergency Response Plan Enhancements Rochester Gas & Electric Corporation R. E. Ginna Nuclear Power Plant Docket No. 50-244

Section 2OS1: Access Control to Radiologically Significant Areas

Procedures

A-1, Revision 66, Radiation Control Manual
A-3, Revision 48, Containment Vessel Access Requirements
A-1.6, Revision 20, Station ALARA Committee
A-1.8, Revision 16, Radiation Work Permits
A-1.1, Revision 40, Access Control to Locked High Radiation and Very High Radiation Areas
A-1.6.1, Revision 27, ALARA Job Reviews,
IP-CAP-1, Revision 16, Abnormal Condition Tracking Initiation or Notification (ACTION) Report
RP-JC-RADIOGRAPH, Revision 9, Support of Radiography Operations
RP-EXP-EXT-LIMIT, Revision 14, Determining External Exposure Control Levels
RP-SUR-POST, Revision 2, Radiological Postings and Boundary Control
RP-SUR-RADIATION, Revision 2, Performance of Radiation Surveys
RP-SUR-PERS-DECON, Revision 18, Personnel Contamination
RP-JC-JOB COVERAGE, Revision 4, Job Coverage
RP-SUR-HOTPART, Revision 0, Performance of Hot Particle Surveys
RP-JC-HOTPART-ASSESS, Revision 8, Hot Particle Dose Assessment
RP-SUR-CONTAM, Revision 0, Performance of Contamination Surveys
RP-ALA-REVIEW, Revision 6, ALARA Job Review Preparation
RP-ALA-SHIELD, Revision 9, Control of Temporary Lead Shielding
RP-SUR-LABEL, Revision 2, Labeling and Control of Radioactive Material
RP-JC-ALARM-PORTAL, Revision 4, Response to Portal Monitor Alarms
RP-JC-AIRSAMPLE, Revision 10, Operation of Portable Air Sampling Equipment

Other

Reactor Vessel Bottom Inspection Overview
Source Term Reduction - Chemistry Initiatives

Section 2OS2: ALARA Planning and Controls

Procedures

A-1, Revision 66, Radiation Control Manual
A-1.6, Revision 20, Station ALARA Committee
A-1.8, Revision 16, Radiation Work Permits
A-1.1, Revision 40, Access Control to Locked High Radiation and Very High Radiation Areas
A-1.6.1, Revision 27, ALARA Job Reviews,
IP-CAP-1, Revision 16, Abnormal Condition Tracking Initiation or Notification (ACTION) Report
RP-JC-RADIOGRAPH, Revision 9, Support of Radiography Operations
RP-EXP-EXT-LIMIT, Revision 14, Determining External Exposure Control Levels
RP-SUR-POST, Revision 2, Radiological Postings and Boundary Control
RP-SUR-RADIATION, Revision 2, Performance of Radiation Surveys
RP-SUR-PERS-DECON, Revision 18, Personnel Contamination
RP-JC-JOB COVERAGE, Revision 4, Job Coverage
RP-SUR-HOTPART, Revision 0, Performance of Hot Particle Surveys
RP-JC-HOTPART-ASSESS, Revision 8, Hot Particle Dose Assessment
RP-SUR-CONTAM, Revision 0, Performance of Contamination Surveys
RP-ALA-REVIEW, Revision 6, ALARA Job Review Preparation
RP-ALA-SHIELD, Revision 9, Control of Temporary Lead Shielding
RP-SUR-LABEL, Revision 2, Labeling and Control of Radioactive Material
RP-JC-ALARM-PORTAL, Revision 4, Response to Portal Monitor Alarms
RP-JC-AIRSAMPLE, Revision 10, Operation of Portable Air Sampling Equipment
RP-TLD-142-10-OPS, Revision 2, Operation of Model 142-10 Panoramic Irradiator
CH-SHUTDOWN-ACTIONS, Rev 6, Chemistry Actions following Plant Shutdown
ALARA PLAN, REACTOR VESSEL CLOSURE HEAD REPLACEMENT, Rev 1
Reactor Head Replacement ALARA Pre-job Analysis, Nos. 030700 & 030701
REFUELING ALARA EXPOSURE ESTIMATES & ANALYSIS, Nos. 030600 -030612
IN-PROGRESS ALARA REVIEW, Reactor Head Replacement
ALARA No. 030603, Head Lift Plan, Rev 1

Other

Chemistry Data for reactor coolant, post shutdown
EPRI-WESTINGHOUSE STANDARD RADIATION MONITORING PROCEDURE FOR
REACTOR COOLANT LOOP PIPING
ALARA COMMITTEE MEETING MINUTES dated June 25, 2003:

Section 3PP2: Access Control

Documents
Safeguards Event Log, June 2002 - July 2003
Security Equipment Testing Procedures

Section 3PP3: Response to Contingency Events

Documents
Security Audit, AINT-2002-011-TGT, September 26, 2002

Section 4OA1: Performance Indicator Verification

Action Reports

2003-2266, 2003-2194, 2003-2120, 2003-2136, 2003-2053, 2003-1850, 2003-1756, 2003-1588, 2003-1431, 2003-1242, 2003-1138, 2003-1001, 2003-0663, 2003-0425

Section 4OA2: Identification and Resolution of Problems

Action Reports
2003-2266, 2003-2194, 2003-2120, 2003-2136, 2003-2053, 2003-1850, 2003-1756, 2003-1588, 2003-1431, 2003-1242, 2003-1138, 2003-1001, 2003-0663, 2003-0425, 2003-1379, 2003–1431, 2003-1331, 2003-1001, 2003-0166, 2003-0086, 2002-2823, 2002-2789, 2002-2431

2002-0717,	White Substance on Jacket of Grey Page Cable
2003-1589	Boron Buildup on V-384A Stem
2003-1599	Boron Buildup on V-868B Stem
2003-1600	Boron Buildup on V-862B Stem
2003-16019	Boron Buildup on V-2224A Stem
2003-1602	Boron Buildup on V-384B Stem
2003-1880	Misinterpretation and Reporting of Siren Status

Documents Reviewed

Reactor Vessel Closure Head Replacement (PCR-0042), Rev. 0
R.E. Ginna Station, Core Exit Thermocouple Nozzle Assembly (EVAL-03-42)
R.E. Ginna Reactor Vessel Head Area Upgrade Package (EVAL-03-23)
Reactor Vessel Closure Head Replacement (5059SCRN 2003-0271)
Ginna Station Part Length CRDM Drive Rod Elimination (EVAL-03-81)
Ginna Nuclear Power Plant CRDM Pressure Housing Assembly Appurtenances ASME III Class 1 Design Report (6 CS 1075)
Framatome ANP Document Submittal - Welder Certifications (FANP-03-2446), dated August 13, 2003
    Procedure Qualification Record 035N005 (for qualifying nameplate to latch housing welding)
Procedure Qualification Record 03SN001 (for qualifying welding of cap-to-rod travel housing)
Weld Procedure Qualification 03SN006 (for welding nameplate to latch housing)
Weld Procedure Qualification 03SN002 (for welding cap-to-rod travel housing)
Weld Procedure Qualification 03SN004 (for welding hatch housing to rod travel housing)
Weld Procedure Specification 76439F1, Rev. 1 (for welding cap-to-rod travel housing)
Weld Procedure Specification 764382F1, Rev. B (for welding nameplate to latch housing)
Pipe Support Inspection Record (for welded RVCH radiation shield pipe support Nos. RVLIS HS-1, RVLIS VS-1, RVH HS-1, PS-276-3-V1, PS-276-4-V2, and RVH VS-1)
SQUA-2003-0003-TJD, Radiation Protection Records
SQUA-2002-0057-HMG, Radiographic Examination

AINT-2002-0006-DHK, Radiation Protection Program
Self-Assessment 2002-0032, Review of RP Instrumentation & Dosimetry Program
Self-Assessment 2002-0049, Review of Contamination Control & Radioactive Materials Control
Performance Indicator Report, June, 2003
SQUA-2003-0068-OAP, Observation of 2003 Refuel Pre-job Briefing
SQUA-2003-0063-OAP, Question Rad worker on RWP
SQUA-2003-0075-OPH, Question Rad worker on radiological conditions
SQUA-2003-0079-DHK, Radiological work practices, removal of protective clothing (PC)
SQUA-2003-0071-OTT, Question Rad worker on RWP
SQUA-2003-0016-OTT, Routine Outage Tour
SQUA-2003-0082-OTT, Question Rad worker on RWP
SQUA-2003-0090-OPH, Question Rad worker on RWP
SQUA-2003-0035-OMS, Rad Con Coaching on PC removal process
SQUA-2003-0038-OMG, Radiation Protection practices in Containment

Drawings

CRDM Ginna Outline for Specification (6MN1191)
Head Materials Drawing (083NA015), Rev. 04, 12 pages
RPV Closure Head Ordering (B&W Canada Drawing 083NE100), Rev. 02
The Arrangement of Reactor Vessel Longitudinal Sections (B&W Drawing 117802E), Rev. 7
Reactor Vessel Head Vent and RVLIS Isometric and Details Drawing (33013-2864), Sheet 1
Support No. RVLIS HS-1 RVLIS Support to Radiation Shield Drawing (10904-0693), Rev. 0
Support No. RVLIS VS-1 RVLIS Support to Radiation Shield Drawing (10904-0696), Rev. 0
Support No. RVH HS-1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0692), Rev. 0
Support No. PS-276-3-V1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0694), Rev. 0
Support No. PS-276-4-V2 Reactor Head Vent Support to Radiation Shield Drawing (10904-0695), Rev. 0
Support No. RVH VS-1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0697), Rev. 0

Procedures

Reactor Head Hydrostatic Test Shop Instruction (259103), Rev. 1
Nondestructive Examination Control (A-903), Rev. 13
Control of Welding (A-901), Rev. 11
Radiographic Technique Sheet for Weld 85 (259123), Rev. 1
Framatome ANP Procedure CRDM Torquing and Welding Reactor Vessel Head Replacement (No. 7 MN 10924), Rev. A
Framatome ANP Nondestructive Examination Procedure Visible Solvent Removable Liquid Penetrant Examination Procedure (54-ISI-240-41), Rev. February 10, 2003
Framatome ANP Welding Components on Reactor Vessel Head Adapters Using Automatic Orbital GTAW Process Welding Machine Type ESAB ProTig 315 (6 MN 11911), Rev. B
Framatome ANP Engineering Verification of Reactor Vessel Closure Head Before and After Seal Welds Machine Type ESAB ProTig 315 (6 MN 1250), Rev. J
EPG-1, Emergency Planning Guideline, Rev 23

Other

ASME Code Form N-2 for the replacement RPV head forging, dated July 17, 2002
ASME Code Form N-2 for the replacement RPV head assembly, dated August 7, 2003
NDE Checklist for Ginna RVCH SN O83N-01, BWC-CONT-083N
RT Report No. 1 for SN 5210497, ref. hole No. 1, weld 85, dated November 14, 2002
RT Report No. 2 for SN 5211403-1, ref. hole Nos. 6, 12, 15, and 30, weld 85, dated February 18, 2003
RG&E Trip Report on the hydrostatic pressure test for the replacement RVCH, dated July 18, 2003
Framatome ANP Weld Control Record, Ginna Replacement RVCH Assembly (Outside Containment) (Process Traveler No. 50-5028903-01), dated August 14, 2003
Framatome ANP Procedure Test Specimen Welding Data Sheet (No. 7 MN 10924), dated August 20, 2003, August 22, 2003, and August 26, 2003
Certificates of Analysis for Argon Bottles (Cylinder Nos. 33-010268, 33-008148, 33-009272, and 33-007132)
Component - adapter welding data sheet and GTAW welding checklist, fit-up before welding, and liquid penetrant examination data for the following adapter/component welds: F12/2610, H4/2590, G13/2593, J3/2606, J13/2607, K4/2608, K8/2588, K12/2609, L5/2583, L11/2585, M6/2589, M8/2586, M10/2584, N7/2587, and N9/2582Analysis of Containment Floor at Elevation 274' 6" (Calc. No. 2060-C-7.1), Rev. 0
Overhead Door and Rotor (EWR No. 2192), Rev. 0
Turbine Building Structural Floor Framing to Support New Reactor Vessel Closure Head Replacement (DA-CE-2003-029), Rev. 0
Containment Building Crane Rigging for Old Head Lift (Calc. No. 2060-C3.3), Rev.1
Runway System (Calc. No. 2060-C4.1), Rev. 1
Containment Building New Head Rigging Test (Calc. No. 2060-C6.1), Rev. 2
Tie-down of Head to Transporter (Calc. No. 2060-C5.2), Rev. 0
Bigge Transporter Configuration, Pull Force and Wheel Loads (Calc. No. 2060-C5.1), Rev. 0
Ginna Turbine Building Runway System (Calc. No. 2060-C4.2), Rev. 1
Apex Plate Girder for HAUP & CRDM Support (Calc. No. 2060-C3.4), Rev.0
Rigging and Gantry/Crane for Loading or Offloading Bigge Transporter (Calc. No. 2060-C2.1), Rev. 1
Action Report (AR) No. 2003-2059
Incident Report 02NX0849 dated June 12, 2002
Receipt Inspection of Reactor Vessel Head Action Report No. 2003-1827
Framatome ANP Nonconformance Report No. 6028485
Framatome ANP Nonconformance Report No. 6028487
Framatome ANP Work Instruction WI-3 Condition Report No. 6028431

Section 4OA5: Other Activities

Documents

NDE-VT-116, Visual Examination of Reactor Vessel Head
Certification Records for Plant and Contractor Personnel
AR 2003-2193, Minor Abrasion on Lower Reactor Vessel Nozzle
AR 2003-2195, Boric Acid Residue on Lower Vessel Head
EPRI Draft Guidance, Sampling and Analysis Guidance for Deposits on Reactor Pressure Vessels at Various Locations
Interoffice Correspondence Between M. Shields and G. Jones, "A" Sump Samples [Samples Taken from the Reactor Vessel Lower Head]

Logical Work Flow of Lower Reactor Vessel Head Visual Examination
RWP #031 061, Survey Maps for Lower Head and ALARA Report
B&W Drawing 117828E, Revision 6, Reactor Vessel Instrumentation Nozzle Details
Procedure Number RF-65.1, Volume II, Section 1.2.12, Reactor Cavity Seal Ring Installation
B&W Drawing 21489-188, Revision 3, Inflatable Reactor Cavity Seal Ring
Project Implementation Plan, Ginna Reactor Vessel Closure Head Replacement
2003 Station ALARA Committee Meeting Minutes
Reactor Vessel Closure Head Replacement (PCR-0042), Rev. 0
R.E. Ginna Station, Core Exit Thermocouple Nozzle Assembly (EVAL-03-42)
R.E. Ginna Reactor Vessel Head Area Upgrade Package (EVAL-03-23)
Reactor Vessel Closure Head Replacement (5059SCRN 2003-0271)
Ginna Station Part Length CRDM Drive Rod Elimination (EVAL-03-81)
Ginna Nuclear Power Plant CRDM Pressure Housing Assembly Appurtenances ASME III Class 1 Design Report (6 CS 1075)
Framatome ANP Document Submittal - Welder Certifications (FANP-03-2446), dated August 13, 2003
Procedure Qualification Record 035N005 (for qualifying nameplate to latch housing welding)
Procedure Qualification Record 03SN001 (for qualifying welding of cap-to-rod travel housing)
Weld Procedure Qualification 03SN006 (for welding nameplate to latch housing)
Weld Procedure Qualification 03SN002 (for welding cap-to-rod travel housing)
Weld Procedure Qualification 03SN004 (for welding hatch housing to rod travel housing)
Weld Procedure Specification 76439F1, Rev. 1 (for welding cap-to-rod travel housing)
Weld Procedure Specification 764382F1, Rev. B (for welding nameplate to latch housing)
Pipe Support Inspection Record (for welded RVCH radiation shield pipe support Nos. RVLIS HS-1, RVLIS VS-1, RVH HS-1, PS-276-3-V1, PS-276-4-V2, and RVH VS-1)

Drawings

CRDM Ginna Outline for Specification (6MN1191)
Head Materials Drawing (083NA015), Rev. 04, 12 pages
RPV Closure Head Ordering (B&W Canada Drawing 083NE100), Rev. 02
The Arrangement of Reactor Vessel Longitudinal Sections (B&W Drawing 117802E), Rev. 7
Reactor Vessel Head Vent and RVLIS Isometric and Details Drawing (33013-2864), Sheet 1
Support No. RVLIS HS-1 RVLIS Support to Radiation Shield Drawing (10904-0693), Rev. 0
Support No. RVLIS VS-1 RVLIS Support to Radiation Shield Drawing (10904-0696), Rev. 0
Support No. RVH HS-1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0692), Rev. 0
Support No. PS-276-3-V1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0694), Rev. 0
Support No. PS-276-4-V2 Reactor Head Vent Support to Radiation Shield Drawing (10904-0695), Rev. 0
Support No. RVH VS-1 Reactor Head Vent Support to Radiation Shield Drawing (10904-0697), Rev. 0

Procedures

Reactor Head Hydrostatic Test Shop Instruction (259103), Rev. 1
Nondestructive Examination Control (A-903), Rev. 13
Control of Welding (A-901), Rev. 11
Radiographic Technique Sheet for Weld 85 (259123), Rev. 1

Framatome ANP Procedure CRDM Torquing and Welding Reactor Vessel Head Replacement (No. 7 MN 10924), Rev. A
Framatome ANP Nondestructive Examination Procedure Visible Solvent Removable Liquid Penetrant Examination Procedure (54-ISI-240-41), Rev. February 10, 2003
Framatome ANP Welding Components on Reactor Vessel Head Adapters Using Automatic Orbital GTAW Process Welding Machine Type ESAB ProTig 315 (6 MN 11911), Rev. B
Framatome ANP Engineering Verification of Reactor Vessel Closure Head Before and After Seal Welds Machine Type ESAB ProTig 315 (6 MN 1250), Rev. J

Other

ASME Code Form N-2 for the replacement RPV head forging, dated July 17, 2002
ASME Code Form N-2 for the replacement RPV head assembly, dated August 7, 2003
NDE Checklist for Ginna RVCH SN O83N-01, BWC-CONT-083N
RT Report No. 1 for SN 5210497, ref. hole No. 1, weld 85, dated November 14, 2002
RT Report No. 2 for SN 5211403-1, ref. hole Nos. 6, 12, 15, and 30, weld 85, dated February 18, 2003
RG&E Trip Report on the hydrostatic pressure test for the replacement RVCH, dated July 18, 2003
Framatome ANP Weld Control Record, Ginna Replacement RVCH Assembly (Outside Containment) (Process Traveler No. 50-5028903-01), dated August 14, 2003
Framatome ANP Procedure Test Specimen Welding Data Sheet (No. 7 MN 10924), dated August 20, 2003, August 22, 2003, and August 26, 2003
Certificates of Analysis for Argon Bottles (Cylinder Nos. 33-010268, 33-008148, 33-009272, and 33-007132)
Component - adapter welding data sheet and GTAW welding checklist, fit-up before welding, and liquid penetrant examination data for the following adapter/component welds: F12/2610, H4/2590, G13/2593, J3/2606, J13/2607, K4/2608, K8/2588, K12/2609, L5/2583, L11/2585, M6/2589, M8/2586, M10/2584, N7/2587, and N9/2582Analysis of Containment Floor at Elevation 274' 6" (Calc. No. 2060-C-7.1), Rev. 0
Overhead Door and Rotor (EWR No. 2192), Rev. 0
Turbine Building Structural Floor Framing to Support New Reactor Vessel Closure Head Replacement (DA-CE-2003-029), Rev. 0
Containment Building Crane Rigging for Old Head Lift (Calc. No. 2060-C3.3), Rev.1
Runway System (Calc. No. 2060-C4.1), Rev. 1
Containment Building New Head Rigging Test (Calc. No. 2060-C6.1), Rev. 2
Tie-down of Head to Transporter (Calc. No. 2060-C5.2), Rev. 0
Bigge Transporter Configuration, Pull Force and Wheel Loads (Calc. No. 2060-C5.1), Rev. 0
Ginna Turbine Building Runway System (Calc. No. 2060-C4.2), Rev. 1
Apex Plate Girder for HAUP & CRDM Support (Calc. No. 2060-C3.4), Rev.0
Rigging and Gantry/Crane for Loading or Offloading Bigge Transporter (Calc. No. 2060-C2.1), Rev. 1
Action Report (AR) No. 2003-2059
Incident Report 02NX0849 dated June 12, 2002
Receipt Inspection of Reactor Vessel Head Action Report No. 2003-1827
Framatome ANP Nonconformance Report No. 6028485
Framatome ANP Nonconformance Report No. 6028487
Framatome ANP Work Instruction WI-3 Condition Report No. 6028431

Section 4OA7: Licensee-Identified Violations

Procedures

RP-TLD-142-10-OPS, Revision 2, Operation of Model 142-10 Panoramic Irradiator