U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-244/85-12

Docket No. 50-244

Licensee No. DPR-18 Priority -- Category C

Licensee: Rochester Gas and Electric Corporation
49 East Avenue
Rochester, New York 14649

Facility Name: R. E. Ginna Nuclear Power Plant

Inspection at: Ontario, New York

Inspection Conducted: June 16, 1985 through July 31, 1985

Inspector: W. A. Cook, Resident Inspector, Ginna

Approved by: L. T. Doerflein for 8/23/85
J. C. Linville, Chief, Reactor Date
Project Section No. 2C, DRP

Inspection Summary:

Inspection on June 16, 1985 through July 31, 1985 (Report No. 50-244/85-12)
Areas Inspected: Routine, onsite, regular, and backshift inspection by the resident inspector (186 hours). Areas inspected included: plant activities during routine power operations; licensee action on previous findings; surveillance testing; plant maintenance; vital battery rack seismic qualification; control rod misposition review; IE Bulletin review; review of spent fuel pool cooling system design; review of Integrated Plant Safety Assessment Review items; Licensee Event Report review; and inspection of accessible portions of the facility during plant tours.

Results: Of the eleven areas inspected, no violations were identified. The receipt of spent fuel shipments from West Valley commenced July 1, 1985, (see paragraph 3.a.). The inspector will review the licensee's analysis of post-outage reactor trips and long-term corrective actions in a subsequent report, (see paragraph 11).

DETAILS

1. Persons Contacted

During this inspection period, the inspector interviewed and talked with operators, technicians, engineering, supervisory and management personnel.

2. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (78-16-05): Flowmeters for IWP Testing not calibrated. During an earlier routine review of Inservice Testing, the inspector identified the use of flowmeter FI-929, Safety Injection Test Line Flow, as not meeting the calibration requirements of Section XI of the ASME Boiler and Pressure Vessel Code. The inspector reviewed the current calibration requirements of FI-929 as specified in Administrative Procedure A-1105, "Calibration and/or Test Surveillance Program for Instrumentation/Equipment of Safety-Related Components" Revision 17, dated 2/27/85. The inspector determined that the A-1105 calibration specifications for FI-929 satisfy the requirements of paragraph IWP-4140 of Section XI of the Code, 1977 Edition, as committed to by the licensee.

Further inquiries by the inspector determined that replacements for FI-929 and a similar type in-line flowmeter FI-933, Containment Spray Test Line Flow, are no longer manufactured by the supplier. Licensee engineering staff is reviewing suitable replacement flowmeters. The inspector will follow licensee actions to procure replacements since currently installed flowmeters were due for replacement in May this year. (85-12-01)

(Closed) Inspector Follow-up Item (78-20-03): Review CS and NaOH Additive System Test Results. During a previous inspection, the inspector was unable to complete the review of monthly test procedure PT-3, "Containment Spray Pumps and NaOH Additive System" because the procedure was not completed by the close of the inspection visit. The inspector determined that although the licensee does not administratively require the retention of completed periodic tests beyond five years, the station central files does maintain many completed procedures on microfilm in excess of this time requirement. The inspector reviewed PT-3, completed on September 28, 1978, and found no discrepancies. In addition, a sampling of PT-3s completed between August 1978 and May 1985 were reviewed with no problems identified.

(Closed) Inspector Follow-up Item (79-BO-3A): Longitudinal weld defects in ASME SA-302 Type 304 stainless steel pipe. The details of the inspector's review of this item are documented in paragraph 8 of this report.

(Closed) Unresolved Item (79-04-03): Review need for revising plant evacuation procedures to accommodate loss of evacuation alarms. During a previous inspection, the inspector determined that the licensee does not

have specific written contingency procedures for the evacuation of station personnel in the event the plant evacuation alarms are disabled due to a loss of the Public Address (PA) system. Further review of this item indicates there are no specific requirements for the development of these contingency plans. The licensee does have established Site Contingency (SC) procedures which provide adequate guidance for the evacuation, accountability, search and rescue of station personnel, if required, in an emergency. The inspector also determined that, at the discretion of the shift supervisor, radio equipped shift personnel and security guard force members would be dispatched to pass the word for a plant evacuation, if necessary. No licensee action is required for this item. This item is resolved.

(Closed) Inspector Follow-up Item (79-06-02): Review RG&E evaluation and corrective actions for snubber MS-146 (top). During an earlier inspection, the inspector identified the presence of grease or fluid along the external shaft of hydraulic snubber MS-146 (top). A review of station records indicate that on October 9, 1979 maintenance procedure M-40, "Surveillance and Maintenance of Hydraulic Snubbers," was completed for MS-146 (top). The snubber was replaced with a rebuilt spare after it was determined that the oil reservoir would not hold oil due to a leak. Subsequent monthly surveillances of the replacement MS-146 (top) snubber were recorded as satisfactory.

(Closed) Inspector Follow-up Item (79-19-02): Re-evaluate 14 day report associated with LER 79-23. On December 7, 1979, during the performance of routine inservice inspection of pressurizer power operated relief valve nozzle-to-safe-end welds, the licensee identified linear cracks near the safe end. As reported in LER 79-023/01T-0, dated December 21, 1979, the investigation efforts to determine the potential causes for the cracking had not been concluded. Original fabrication contamination or chlorides leaching from the thermal insulation were suspected as possible causes. Pressurizer nozzle insulation material was replaced while awaiting further analysis results.

Follow-up LER 79-023/01X-1, dated August 5, 1980, concluded from a May 29, 1980 Metallurgical Analysis, that the cracks were caused by super solidus cracking or "hot cracking" during original fabrication as the molten weld metal cooled to the solid state. Inadequate delta ferrite content of the stainless steel weld metal at the weld interface and a geometric discontinuity at the nozzle carbon steel/weld metal interface were identified as the main contributors to the cracking process. This type of cracking was particular to one local weld area and was not considered a generic concern by the licensee. Reexamination by the licensee during the subsequent 1980 refueling and maintenance outage found no further crack indications.

(Closed) Inspector Follow-up Item (79-BU-18): Audibility problems encountered on evacuation of personnel from high-noise areas. The details of the inspector's review of this item are documented in paragraph 8 of this report.

(Closed) Inspector Follow-up Item (83-17-02): Revise 'E' procedures to better address loss of service water. NUREG-0821, Integrated Plant Safety Assessment Systematic Evaluation Program Report for R. E. Ginna Nuclear Power Plant, dated December 1982, specified in item 3.3.1.1 that the licensee had agreed to install a temporary hose connection from the station's city water firemain to the Emergency Diesel Generators (EDG) cooling water systems as an alternate source in the event of a loss of service water. The modification was reviewed and documented in an earlier inspection (Report No. 50-244/83-17), however, licensee's emergency procedures were not revised to reflect this system modification. The inspector reviewed Emergency Procedure (E)-38, "Loss of Service Water," Revision 6, dated 8-22-84, and Attachment A-0.1 to E-4.3, "Loss of All A.C. Power," Revision 8, dated 4-4-85, and verified that adequate procedural guidance is provided for the line-up of alternate cooling to the EDGs upon the loss of station service water.

(Closed) Inspector Follow-up Item (85-02-01): Review completed DC power bus modifications. Vital battery charge/discharge flow monitoring units, installed as a result of a NUREG-0821, "Integrated Plant Safety Assessment Review" imposed upgrade, were found with meters which were grounded to the unit chassis. When energized, these meters would overload and short. The contractor replaced the original Weston amp meters with new Analogic meters and, with the exception of a circuit board fault identified and corrected in the 'B' battery monitors, tested them satisfactorily on May 23, 1985. The inspector reviewed completed Station Modification Procedure SM-3199.3, "Vital Battery Load Flow Monitor Rework," Revision 0, dated January 9, 1985 and associated surveillance reports, nonconformance reports, change notices and licensee correspondences. No discrepancies were found.

The inspector had no further questions.

3. Review of Plant Operations

a. Throughout the reporting period, the inspector reviewed routine plant operations. The reactor has been operating at full power since June 8, 1985 following a reactor trip on June 6 which was reviewed and documented in Inspection Report No. 50-244/85-10. On July 1 and 2, 1985, the licensee received its first two spent fuel assembly shipments from the U.S. Department of Energy, West Valley Demonstration Project in West Valley, New York. Observation of licensee receipt inspection and radiation/contamination surveys and verification by independent measurement were conducted by a region-based inspector and the resident inspector. No descrepancies were noted. Transfer of the spent fuel assemblies to the Spent Fuel Pool (SFP) is pending completion of the Auxiliary Building crane modifications.

On July 17, 1985, the licensee conducted a Safeguards Contingency Plan Exercise.

Members of the New York State Radiological Emergency Preparedness Group, the Wayne County Emergency management and the Federal Bureau of Investigation, as well as, the NRC resident inspector were observers of the drill. The exercise was developed by the NYS Emergency Management Agency with the assistence of other drill participants. The exercise was designed to test command/control, communications and response force mobilization.

b. During the inspection, the inspector toured accessible plant areas. Items reviewed include radiation protection controls, plant housekeeping, fire protection, equipment tagging and security.

c. Inspector tours of the control room this inspection period included review of shift manning, operating logs and records, and equipment and monitoring instrumentation status.

d. Safety system valves and electrical breakers were verified to be in the position or condition required for the applicable plant mode as specified by Technical Specifications and plant lineup procedures. This verification included routine control board indication review and conduct of a partial systems lineup check of the Standby Auxiliary Feedwater System on June 27 and July 3, and the Containment Spray System on July 26.

4. Surveillance Testing

a. The inspector witnessed the performance of surveillance testing of selected components to verify that the test procedure was properly approved and adequately detailed to assure performance of a satisfactory surveillance; test instrumentation required by the procedure was calibrated and in use; the test was performed by qualified personnel; and the test results satisfied Technical Specifications and procedural acceptance criteria, or were properly resolved.

b. The inspector witnessed the performance of portions of the following tests:

Periodic Test, (PT)-2.1, "Safety Injection System Pumps," performed on June 17, 1985.

PT-2.3, "Safeguard Valve Operation," performed on June 24, 1985.

PT-2.8, "Component Cooling Pump System," performed on July 8, 1985.

5. Plant Maintenance

a. During the inspection period, the inspector observed maintenance and problem investigation activities to verify compliance with regulatory requirements; compliance with administrative and maintenance

procedures; required QA/QC involvement; proper use of safety tags; proper equipment alignment and use of jumpers; personnel qualifications; radiological controls for workers protection; and ascertain reportability as required by Technical Specifications.

b. The inspector witnessed the following maintenance activity:

-- The performance of minor maintenance on the 1C Standby Auxiliary Feedwater Pump in accordance with Maintenance Procedure (M)-11.14, "Inspection and Maintenance of Ingersol Rand Pumps," Revision 12, December 21, 1983, performed on July 11, 1985.

The inspector found no discrepancies.

6. Review of Vital Battery Racks Seismic Qualification

References:

NUREG/CR-1821, "Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program," dated December 1980

Ginna Station Design Analysis, "Battery Rack Horizontal Restraint," EWR No. 2831, Revision 0, Dated May 11, 1981

Ginna Station Design Criteria, "A and B Vital Battery System Replacement," EWR No. 3891, Revision 1, dated January 15, 1985

Jordan to Starostecki Memo, dated February 8, 1985

GNB Batteries, Inc. to Rochester Gas & Electric Corp. letter, dated March 27, 1985.

RG&E Drawing No. 33013-1120, Revision 5, dated April 12, 1985

RG&E Drawing No. 21489-492, Revision 3, dated April 12, 1985

A recent analysis by GNB Batteries, Inc. indicates that battery racks and batteries at some nuclear power plants may not have been installed in accordance with initial seismic qualification testing specifications. The potential problem involves the placement of battery cells on the rack with an excessive gap between the end cells and the battery rack side stringers or restraints. Seismic qualification testing specified that this end gap be no greater than one-quarter inch.

The originally designed and installed battery racks at Ginna Station were manufactured by GNB Batteries, Inc.. The inspector determined that the vital battery racks were modified in 1981 after completion of a 1980 seismic review conducted in conjunction with the Systematic Evaluation Program (SEP). The seismic review team concluded that the battery rack seismic design was adequate with the exception of the wooden lateral bracing. The team recommendation was for the replacement or strengthening

of these restraints to ensure their ability to carry full seismic inertia loading. The 1981 battery rack modification involved replacement of wooden restraints with seismically braced structural steel restraints and a reanalysis of the rack seismic qualifications. In addition, foam block material was used to further prevent cell displacement in the rack and prevent impact loading.

During the 1985 Annual Refueling and Maintenance Outage the licensee further modified the vital battery systems. The original Gould FTA-19 storage cells in the 'A' vital battery bank were replaced with new GNB NAX 1200 storage cells without modification of the existing battery storage racks. (The 'B' vital battery storage cells are scheduled to be replaced in 1986). The new NAX 1200 cells are approximately the same weight as the old FTA-19 cells, however, the new storage cells have a smaller dimensional width. Consequently, additional foam spacing material is used to compensate for the larger gaps between the storage cells and the rack end stringers.

The inspector determined that the licensee received notification of the potential battery bank end gap problems concurrent with the 1985 modification work. GNB Batteries, Inc. recommendations were to provide spacers of either wood, open cell styrofoam or comparable GNB approved materials to fill the end gaps. Since the licensee was already using an approved foam (Armaflex) padding material for spacing purposes and had previously analyzed its use, the battery racks appear to satisfy initial seismic qualification requirements.

7. Control Rod Mispositioning

IE Information Notice No. 83-75 and Institute of Nuclear Power Operations (INPO) Significant Operating Experience Report (SOER) No. 84-2 identified industry problems regarding the mispositioning of reactor control rods. The inspector reviewed the licensee's response to this safety issue to determine if additional NRC action is warranted and to determine the extent of the licensee's response to non-regulatory INPO recommendations.

The inspector reviewed Station Emergency Procedures E-7, "Drop of a Rod Cluster Control Assembly," E-10, "Malfunctioning Rod Position Indicator," and E-12.1, "Malfunctioning Control Rod" and determined that adequate procedural guidance and precautions are provided for the proper control of, or the recovery from, a mispositioned control rod assembly. Compensatory measures for the loss of rod position indications are also clearly defined.

In addition, the inspector interviewed licensed operators and reviewed the associated Training Department lesson text covering this safety issue. Station operators appear to have been properly appraised of the recent industry events and are aware of the potential consequences of improper rod positioning. The lesson text included a comprehensive review of applicable station procedures and Technical Specifications governing control rod mispositioning and rod position indication malfunctions.

The inspector had no further questions.

8. IE Bulletin Follow-up

The inspector reviewed licensee actions on the following IE Bulletins to determine that the written response was submitted within the required time period, that the response included the information required including adequate corrective action commitments, and that licensee management provided adequate dissemination of the bulletin and the response. The review included discussions with licensee personnel and observations of the items discussed below.

79-03 and 79-03A: Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe. IE Bulletin No. 79-03 required the licensee to determine if ASME SA-312, Type 304 stainless steel pipe manufactured by Youngstown Welding and Engineering Company was in use or planned for use in safety-related systems. Subsequent findings, addressed in IE Bulletin No. 79-03A, indicated that ASME Code NDE requirements were not adequate for the detection of lack of weld centerline penetration (CLP). As a result, any manufacturer's SA-312, Type 300 series austenitic stainless steel pipe fabricated by use of longitudinal fusion welds may contain undetected CLP. All licensees with this type of pipe installed were required to identify its specific applications and provide the NRC staff with information relating to the structural integrity of the piping components.

The inspector reviewed the licensee's responses documented in White to Grier letters, dated April 6, 1979 and August 5, 1980. Class 2 stainless steel piping systems at Ginna Station are made from seamless pipe which was originally designed to ASA-B36.19 and B36.10 with reference in applicable line specifications to ASTM-A-312, Type 304 stainless steel seamless pipe. The licensee did conduct a comprehensive review of safety-related stainless steel piping systems and identified one fitting in the residual heat removal system with longitudinal seam welds. The inspector concurred with the licensee's assessment that this fitting did not fall within the category of concern addressed by IE Bulletins 79-03 and 79-03A. These bulletins are closed.

79-18: Audibility Problems Encountered on Evacuation of Personnel from High-noise Areas. This IE bulletin required licensees to determine whether current alarm systems and evacuation announcement systems are clearly audible and/or visible throughout all plant areas, with emphasis on high-noise areas. In addition, the IE bulletin required that corrective action be taken to assure inaudible areas receive adequate audible/visual evacuation alarms or that additional compensatory administrative measures be instituted to assure personnel evacuation.

The inspector reviewed the licensee's response documented in White to Grier letter, dated September 21, 1979, and determined that there were no discrepancies. Accessible high-noise areas identified as requiring upgrading were reviewed by the inspector and the visual alarm placements were found satisfactory. For those areas where additional audible

or visual alarms were not used, the inspector determined that compensatory measures were adequate. This bulletin is closed.

9. Review of Spent Fuel Pit Cooling System

The inspector conducted a review of the Spent Fuel Pit (SFP) Cooling System to determine if the system design was susceptible to siphoning, potential lowering of SFP water level and uncovering spent fuel assemblies. A review of system drawings, a system walkdown verification and a review of system operating and maintenance procedures indicated that essential design features and operating guidelines have been implemented to prevent an inadvertent SFP drain down.

Although the SFP cooling pump discharge piping discharges near the bottom of the SFP storage racks, a 1/4 inch diameter drilled hole in the discharge piping, located approximately 18 inches below the normal SFP water level, provides an anti-siphoning effect. In addition, SFP water level alarms received in the Control Room would alert operators of an abnormal level change. SFP cooling pump suctions are located 24 inches below normal water level and 5 feet 8 inches above the top of the spent fuel racks. These penetrations preclude the possibility of draining the pool and assure a minimum of 5 feet 8 inches of water above the spent fuel.

As discussed in Inspection Report 50-244/85-06, paragraph 6, the only other possible means of lowering the SFP water level would be via a failure of the reactor cavity seal while the transfer canal gate valve was open with no operator intervention. Even in this worst case scenario, approximately one foot of water would still remain to cover the spent fuel assemblies in the SFP storage racks.

The inspector had no further questions.

10. Integrated Plant Safety Assessment Review

The inspector reviewed the below listed items which were identified during the Systematic Evaluation Program (SEP). The SEP reviews required equipment and/or procedural changes. The topic numbers refer to paragraph designations in NUREG-0821, "Integrated Plant Safety Assessment Final Report for the R. E. Ginna Nuclear Power Plant," December 1982.

a. Response-Time Testing of Engineered Safety Features (3.3.6.1). SEP topic VI-10.A reviewed the testability and operability of the reactor protection and Engineered Safety Features (ESF) systems. Testing these systems is necessary to demonstrate a high degree of availability and to show that the response times assumed in accident analyses are within the design specifications. Initial NRC staff review showed that the instrumentation strings from sensors through bistable devices were not response-time tested by the licensee. In a November 5, 1980 letter in response to TMI Lessons Learned, the licensee agreed to implement a response-time testing program for all

instrument strings that initiate auxiliary feedwater and containment isolation.

Response-time testing of instrument channels associated with containment isolation and automatic auxiliary feedwater actuation is conducted in accordance with Station Periodic Test (PT)-32.3, "Containment Isolation and Auxiliary Feedwater System Response-Time Verification Test". The inspector reviewed all completed PT-32.3 procedures to verify: response-time testing was conducted at the specified surveillance interval; all actuation channels were periodically tested; the test procedure is clearly and adequately written; and that test results are within Technical Specification acceptance criteria. The inspector found no discrepancies.

b. Steam Generator Blowdown System (4.12.3). NRC Staff review of the steam generator (S/G) blowdown system, with respect to missile generation and protection, was completed and found acceptable except for the valve operator of containment isolation valve No. 5738. The failure of valve No. 5738 could generate a missile that could potentially have an effect on nearby safety-related components. The licensee committed to install a restraint on the operator to eliminate this potential problem.

The modification to valve No. 5738 (1A S/G blowdown line containment isolation valve) was accomplished under Station Modification Procedure (SM)-2512.78, "Seismic Upgrade of BDU-28 on Analysis Line SGB-300 S/G Blowdown in Intermediate Building," Revision 0, dated March 7, 1984. The inspector verified the proper installation of the valve operator restraint and reviewed the completed modification procedure and associated contractor work procedure, material requisitions, and Quality Control surveillance reports; and found no discrepancies.

c. Essential Service Water Pump Operability (4.15.3). Based on SEP review criteria NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants" and 10 CFR 50, Appendix A, General Design Criterion 2, the NRC staff concluded that the essential service water pumps were not qualified with regard to structural and functional integrity. The lack of seismic support near the suction of the pumps could result in overstressing the pump casing support and potential pump failure. The licensee committed to perform a structural upgrade by June 30, 1984.

The inspector reviewed the completed station modification package, SM-3316.1, "Service Water Pumps Structural Upgrade," and found no discrepancies. The modification consisted of installing two seismic lateral supports on the submerged section of the 14 inch pipe column just above the pump suction bowl assembly. The two supports extend at a 90 degree angle and attach to the wall of the service water bay. The modification was completed for all four service water pumps by

May 20, 1984, and the modification was approved by the PORC on June 20, 1984.

d. Containment Isolation System (4.22). NRC staff review of isolation valves in lines penetrating the containment determined that the following penetrations required additional administrative controls and/or testing:

Penetration 121 is the nitrogen supply to the Pressurizer Relief Tank (PRT) line. The licensee agreed to lock-closed manual isolation valve 547 and leak test it. The inspector verified valve 547 is locked-closed and controlled via Administrative Procedure (A)-52.2, "Control of Locked Valve and Breaker Operation," Revision 63, March 13, 1985, and leak tested in accordance with Periodic Test (PT)-23.2, "Containment Isolation Valve Leak Rate Testing Nitrogen Supply to PRT."

Penetration 129 is the nitrogen supply to the Reactor Coolant Drain Tank (RCDT) line. The licensee agreed to lock-closed manual isolation valve 1793 and leak test it. The inspector verified valve 1793 is locked-closed and controlled via A-52.2, and leak tested in accordance with PT-23.20, "Containment Isolation Valve Leak Rate Testing RCDT Gas Header."

Penetrations 301 and 302 are the containment auxiliary steam heating lines. The licensee agreed to lock-closed manual valves 6152 and 6165 and leak test both valves. The inspector verified both valves are locked closed and controlled via A-52.2, and leak tested in accordance with PT-23.40, "Containment Isolation Valve Leak Rate Testing Auxiliary Steam Supply and Condensate Return."

The inspector reviewed the completed Periodic Test procedures and results for the above stated penetrations back through 1980. No discrepancies were noted.

11. Licensee Event Report (LERs)

The inspector reviewed the following LERs to verify that the details of the event were clearly reported, the description of the cause was accurate, and adequate corrective action was taken. The inspector also determined whether further information was required, and whether generic implications were involved. The inspector further verified that the reporting requirements of Technical Specifications and station administrative and operating procedures had been met, that the event was reviewed by the Plant Operations Review Committee and that continued operation of the facility was conducted within the Technical Specification limits.

85-13: "Manual Actuation of Engineered Safety Feature." At 8:30 A.M. on May 31, 1985, a tornado warning was declared by the U.S. Weather Bureau for the general area of Wayne County. Ginna Station control room

personnel were informed of the potential adverse weather conditions and in accordance with station procedures started the 'A' Emergency Diesel Generator (EDG) and tied it to its respective safeguards bus (Bus 18). The tornado warning was rescinded at 9:24 P.M. that evening, and the 'A' EDG was returned to its standby status, and the Bus 18 normal electrical line-up was restored. The inspector reviewed this event and the subsequent written report, and found no discrepancies.

85-14: "Automatic Actuation of the Reactor Protection System (RPS)." On June 6, 1985, an automatic reactor trip and subsequent turbine trip occurred as a result of an apparent "Overpower Delta Temperature" trip condition. The Overpower Delta Temperature trip actuation was caused by grounding an energized lead on the excore nuclear power range channel N-41 selector switch during its repair. The ground caused an undervoltage spike on Instrument Bus 1D and momentarily deenergized the Overpower Delta Temperature bistable of power range channel N-44. With the channel N-41 Overpower Delta Temperature bistable already tripped because the channel was defeated for repairs, the 2 of 4 trip logic was satisfied and the reactor tripped as designed.

The inspector reviewed this event in the previous inspection period as documented in Inspection Report No. 50-244/85-10, paragraph 3.a. The inspector discussed corrective actions with the licensee and expressed a concern for appropriate management controls. The licensee is analyzing this event and multiple reactor trips which occurred during the recent outage startup to determine where improvements can be made. The inspector will review the results of the licensee's analysis in a subsequent report. (85-12-02)

85-15: "Any Operation or Condition Prohibited by the Plant's Technical Specifications". On June 20, 1985, a non-licensed Auxiliary Operator vented both level transmitters, LI-171 and LI-106, on the 'B' Boric Acid Storage Tank (BAST) contrary to the specific instructions given him to vent only the LI-106 transmitter. Unaware of the consequences of venting both detectors, the Auxiliary Operator's actions resulted in a coincident 2 of 2 logic for the 'B' BAST low level of 10% and caused the automatic realignment of the Safety Injection (SI) pumps suction from the BASTs to the Refueling Water Storage Tank. Control room operators were closely observing BAST level indications during this evolution, identified the problem and quickly reset the low level logic and restored the normal valve line-up within 45 seconds. For this period of approximately 45 seconds, the normal flow paths from the BASTs to the reactor coolant system, via the SI pumps, were not available contrary to Technical Specification 3.2.2.d requirements.

The inspector reviewed this event and discussed with the licensee the corrective actions taken and planned. Those corrective actions include training of all operators on the lessons learned from this event, turnover of all future BAST level detector venting operations to the Instrumentation and Control Technicians, and termination of employment of the Auxiliary Operator involved.

A Notice of Violation is not issued in response to this event, in that: the licensee identified and promptly reported the Techncial Specification (TS) violation, the violation has minimal safety significance, the control room operators were alert and took prompt corrective actions, licensee management has taken appropriate corrective measures to prevent recurrence, and this TS violation is unlike any previous event for which corrective actions could have reasonably been expected to prevent its occurrence.

12. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specification 6.9.1 and 6.9.3 were reviewed by the inspector. This review included the following considerations: the reports contained the information required to be reported by NRC requirements; test results and/or supporting information were consistent with design predictions and performance specifications; and the validity of the reported information. Within this scope, the following report was reviewed by the inspector:

-- Monthly Operating Report for June 1985.

13. Exit Interview

At periodic intervals during the course of the inspection, meetings were held with senior facility management to discuss the inspection, scope and findings.

Based on the NRC Region I review of this report and discussion held with licensee representatives on July 31, 1985, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.