U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-155/89002(DRP)

Docket No. 50-155 License No. DPR-6

Licensee: Consumers Power Company
1945 West Parnall Road
Jackson, MI 49201

Facility Name: Big Rock Point Nuclear Plant

Inspection At: Charlevoix, Michigan

Inspection Conducted: January 1 through February 11, 1989

Inspectors: E. Plettner

N. Williamsen

Approved By: I. N. Jackiw, Chief, Reactor Projects Section 2B

I N Jackiw

2-27-89
Date

Inspection Summary

Inspection on January 1 through February 11, 1989 (Report No. 50-155/89002(DRP))
Areas Inspected: The inspection was routine, unannounced, and conducted by the Senior Resident Inspector and the Resident Inspector, with inputs from regional inspectors. The functional areas inspected consisted of the following: licensee actions on previous inspection findings; surveillance activities; maintenance activities on various components; operational safety verification including fuse replacement resulting in a Notification of Unusual Event; an engineered safety feature walkdown of the liquid poison system; review of the metallurgical report on Target Rock valve failure; NRC Information Notice followup on cracks in shroud support access hole cover welds; NRC Temporary Instructions followup on scram discharge volume capability, motor-operated valve common mode failure during plant transients, fastener testing, and proper handling of emergency diesel generator fuel oil.
Results: The licensee has demonstrated a desire to respond in a timely manner to issues and concerns presented to them by the NRC. The surveillance, maintenance, and operational safety programs appeared to be performed in a manner to ensure public health and safety. The engineered safety feature system walkdown identified no safety concerns. The review of the metallurgical report of Target Rock valve failure was acceptable and the conclusions valid. No significant safety items were identified in this report.

DETAILS

1. Persons Contacted

*T. Elward, Plant Manager
*L. Monshor, Quality Assurance Superintendent
*H. Hoffman, Maintenance Superintendent
*R. Burdette, Acting Chemistry/Health Physics Supervisor
*W. Trubilowicz, Operations Supervisor
*G. Withrow, Plant Engineering Supervisor
*R. Alexander, Technical Engineer
*E. Zienert, Director Human Resources
*P. Donnelly, Nuclear Assurance Administrator
*R. Buckner, Nuclear Training Administrator

The inspectors also contacted other licensee personnel in the Operations, Maintenance, Engineering, Radiation Protection, and Technical Departments.

*Denotes those present at exit interview.

2. Licensee Action On Previous Inspection Findings

(Closed) Violation (155/85022-03): The licensee failed to request an exemption from the requirements of Section III.G.2 of Appendix R after determining that the fire protection features in the Screenwell and Pumphouse did not meet the specific requirements of Section III.G.2, in that no fire suppression system was installed.

Subsequently, by letter dated July 1, 1986, the licensee filed an exemption request. However, upon further licensee review, a more desirable method of shutdown was determined which resulted in a revised exemption request being submitted by letter dated October 14, 1986. This method does not rely on any of the equipment in the Screenhouse and was considered a more conservative method. However, this method involves the temporary attachment of a hose to maintain hot shutdown. This action was considered a repair by Appendix R guidelines; therefore, an exemption was still viewed as appropriate.

During this inspection, the inspectors reviewed the pending exemption correspondence and toured the plant areas where the hose connections would need to be made.

The inspectors provided the following comments regarding the October 14, 1986 licensee exemption request:

a. The Standby Diesel Generator should be identified as the power source for the Demineralized Water System (DWS) pump. It should also identify that additional fuel from off-site sources may be required to meet the postulated 72-hour loss of off-site power conditions.

b. The statement that the flow path from the DWS pump to the Emergency Condenser (EC) is opened by operating an air-operated valve should be corrected to indicate that two valves must be opened.

c. The statement that analyses have been performed to show that when the DWS is connected to the instrument air compressor cooling system, as described in the exemption request, there would be enough cooling water available onsite to permit the EC to operate for a minimum of 36 hours should be qualified by mentioning that this time was arrived at by taking credit for the average of operator log readings for other tanks onsite. There are minimum levels set by Administrative Procedures but no Technical Specification requirements exist for any tank levels.

The inspectors provided these comments to the NRR Licensing Project Manager following the March 1988 inspection visit.

This item was remaining open pending an NRR determination of the acceptability of the licensee's October 14, 1986, submittal; however, since no further regional action is deemed appropriate at this time, this item is closed.

<u>(Closed) Open Item (155/85022-08)</u>: Fire detectors were not installed throughout containment as per NFPA codes. Nor was an arrangement of fire detectors throughout containment observed that would provide prompt detection of incipient fires.

On September 28, 1982, the licensee requested an exemption from having to install a fixed fire suppression system (excluding the recirculation pump room) inside containment. Additionally, the licensee described the fire detection systems installed in specific locations within containment. These locations included the core spray pump room (actually located in a separate room outside containment), control rod drive accumulator area, shutdown heat exchanger room, and the interior cable spreading area.

The NRC granted this exemption by letter dated March 8, 1983, in a Safety Evaluation Report (SER). The wording used in the SER specified that several modifications had been made inside containment to allow rapid fire detection action, that the installed early warning detection system would provide prompt detection of incipient fire conditions, and that fire hoses were distributed throughout containment. With regard to the granted exemptions, the inspector had determined that the installed warning detection system that would provide the referenced prompt detection was accurately detailed in the licensee submittal. To determine the adequacy of the installed detection system, the inspector had reviewed all available historical correspondence between the licensee and NRC which provided a description of the fire detection systems for containment. The licensee's submittals included the July 14, 1978, December 8, 1978, and August 31, 1979 letters. These letters provided NRC requested information and formed the licensing basis for NRC to determine that the as-installed fire detection systems satisfied the criteria of Appendix A to NRC Branch

Technical Position 9.5-1. The NRC transmittals accepting this design and providing background information were dated November 20, 1978 (request for information only), April 4, 1979, and December 17, 1979.

More recently, the licensee has performed engineering analyses to meet Generic Letter 86-10 guidelines so as to justify Appendix R compliance for the installed system configurations. A review of this analysis, together with either a visual inspection of the accessible installed detection systems or a review of drawings for systems that weren't accessible resulted in the inspector concluding that these systems met NRC requirements. Therefore, this portion of the item was considered closed. The additional questions raised during the March 1988 inspection visit are being addressed through the followup of other items. Further, as part of this review, the inspectors, during plant tours, verified that the accumulation of transient combustibles inside containment were being properly maintained, as described in the approved exemptions dated March 8, 1983 and March 26, 1985. This control of combustibles was considered an improvement since the previous Appendix R inspections.

<u>(Closed) Open Item (155/85022-11)</u>: As a result of Appendix R modifications and the upgrading of certain fire protection features which were detailed in the licensee's Fire Protection Program Evaluation document dated March 29, 1977, the inspectors requested the licensee to update this document so as to reflect the present plant fire protection features. In addition, the licensee committed to providing a comprehensive "Summary of Fire Protection Provisions" (FPPSD) document, as described in the licensee's April 14, 1986 transmittal.

By letter dated February 27, 1987, the licensee submitted a FPPSD providing a more detailed up-to-date description of the fire protection and safe shutdown features of the plant. However, as a result of the inspectors review, certain fire protection feature discrepancies were identified, and a more detailed electrical analysis section relative to Appendix R still appeared necessary. Therefore, further revision of this document was required.

The inspectors provided the licensee's staff with examples of identified discrepancies (e.g., Section I, Fire Area 15 (Machine Shop); Paragraph B - specifies that all walls are rated at three hours; however, according to the July 14, 1978 licensee letter, the east wall is rated at two hours) which the licensee acknowledged. The licensee indicated that plans were being made to correct and improve the FPPSD in the areas mentioned. This item remained open pending further review of the licensee's actions to correct the discrepancies and improve the FPPSD.

By letter dated December 7, 1988, the licensee indicated the following:

Section I of the FPPSD has been corrected to accurately describe the barriers in Fire Area 15. Section III, Appendix B, Post Fire Safe Shutdown Methods has been revised to add Fire Area 5, Electrical Equipment Room to the description for Method I. Fire Areas 16 (Turbine Lube Oil Room) and 20 (Pipe Tunnel) were also deleted from Method V description.

Other additions and changes have been completed with additional revisions in progress aimed at improving the overall completeness of the document. The electrical analysis section will be revised following completion of the breaker coordination and High/Low Pressure Interface reviews.

Based on the above licensee submitted information, this item is closed.

(Closed) Open Item (155/88006-04): Required analysis by licensee to show that containment temperature and radioactivity levels are acceptable for entry under postulated Appendix R conditions. This open item deals with the requirement that the licensee analyze containment temperature and radioactivity levels during postulated Appendix R conditions in order to show that personnel entries into containment can be made after a scram and after a safety relief valve has lifted and receded. Under these postulated Appendix R conditions, entry to containment would be needed after a scram and before the end of the 72-hour loss of off-site power period, since there are several actions needed to achieve cold shutdown which require containment entry. Region III staff and Brookhaven National Laboratory inspectors reviewed an analysis by the licensee showing that temperature and radiation levels 16 hours after a relief valve has lifted and reseated will not prevent workers from entering containment. The licensee analysis adequately resolved the issue. This open item is closed.

(Closed) Open Item (155/88006-05a): A review of the shutdown methods described in EMP-3.10 resulted in several recommendations by the inspectors to improve this procedure.

By letter dated December 7, 1988, the licensee provided a revised EMP-3.10 which, as specified, included the inspector's recommendations. Also, according to the December 7, 1988 letter, copies of SOP-28, SOP-5 and plant P&IDs have been placed in the Alternate Shutdown Building (ASB). The licensee also indicated that the above documents have also been added to the annual inventory list to insure they remain available in the ASB.

Based on the above actions, this item is closed.

(Closed) Open Item (155/88006-05b): Additional recommendations to improve EMP-3.10 were suggested.

These recommendations were incorporated as specified in the licensee's December 7, 1988 letter discussed above (155/88006-05a). Based on those actions, this item is closed.

(Closed) Open Item 155/88006-05c): Emergency Procedures Covering Loss of Primary Coolant System (PCS) Make-up Capability, per Appendix R. This open item deals with postulated fire damage in the Control Rod Drive Pump Room, the Shutdown Heat Exchanger Room, and the Reactor Cooling Water Pump Room, and the possible need for additional training of operators or for additional procedures covering loss of PCS make-up capability. The licensee has since issued the Emergency Operating Procedures (EOPs) and has completed operator training. EOP-01, "Primary System Control", Revision 4, dated June 24, 1988, adequately addresses all required actions to mitigate a Loss of Primary Coolant System Make-up event. This open item is closed.

(Closed) Open Item (155/88006-05d): Normal shutdown on loss of off-site power under Appendix R conditions. This open item deals with the possible need for additional procedures to safely respond to fires in areas that do not have automatic detection and which may contain safe shutdown equipment, both with and without off-site power. The licensee responded that a review of existing procedures was performed which concluded that no additional procedures were warranted, since the the EOPs, EMP 3.10, and ONP 2.36, "Loss of Station Power", Revision 118 dated March 23, 1988, are adequate to insure that appropriate operator actions are taken. This open item is closed.

(Closed) Open Item (155/88006-5e): A revision of SOP-28 was recommended to add the Shutdown Cooling Water Pumps as a potential load for either the Standby Diesel Generator or the Emergency Diesel Generator (Step 6.1.3-17).

According to the December 7, 1988 letter, SOP-28 has been revised and issued to address the above concern. Based on this action, this item is closed.

(Closed) Open Item (155/88006-06): A review of training records showed that a Shift Supervisor needed to make up missed BWP-07 training.

As specified in the licensee's December 7, 1988 letter, the required training has been completed. Based on the Shift Supervisor completing this training, this item is closed.

(Closed) Open Item (155/88006-08): The prime means of communication to implement EMP-3.10 is the two-way radio system. A test was performed between the ASB and the Control Room. The results were satisfactory, although the licensee stated that communication between the Containment and the ASB were usually only possible by stepping outside the ASB. The actions inside containment are those necessary to achieve cold shutdown which would occur several hours into the fire scenarios, and do not absolutely require radio communications. This was considered acceptable.

However, since the radio system requires charging after approximately eight hours, and the plant may still be in hot shutdown at that time when communications may still be required, the licensee was asked to show that the radio system could be maintained charged and available for the entire time needed.

As specified by the licensee's letter dated December 7, 1988, numerous charged spare batteries are maintained by the Operations and Security Departments. A revision to EMP-3.10 (Attachment 2) Precaution - Item 4 has been issued to advise operators that spare batteries are available should the need arise. Based on this licensee submitted information, this item is closed.

(Closed Unresolved Item (155/86010-01): Corrective action on issuance of expired procedures. This item deals with a series of five events identified by the licensee where non-licensed personnel apparently failed to follow administrative procedures. The item was unresolved pending NRC review of the effectiveness of corrective actions. Violation No. 155/88013-02 was issued in Inspection Report 155/88013, dated August 5, 1988, for failure to follow administrative procedures. This unresolved item is closed.

(Closed) Open Item (155/86010-02): Possible Training Weaknesses in the Administrative Controls Area. This open item deals with the adequacy of corrective actions including training, for six licensee Deviation Reports and one voluntary Licensee Event Report. These reports involved apparent failures to follow administrative procedures. The inspector reviewed the Reports and concluded that the corrective actions, many of which included training, were adequate. This item is closed.

The following items have been closed during this inspection period based on a directive by the Division Director, Division of Reactor Safety, Region III. Our decision to close these items was based on the length of time the item has been in existence and the recognition of limited safety significance.

(Closed) Open Item (155/86013-02): Items Removed from the MEL Still Relied on in Emergency - EQ Procedures.

(Closed) Open Item (155/86013-03): Use of Inaccurate Dates in PACS Listing of Maintenance and Surveillance Activities- EQ.

(Closed) Open Item (155/86013-04): EQ Awareness Training for Plant Personnel.

(Closed) Unresolved Item (155/86013-07): Flow Transmitter FT-2162 not Qualified Prior to EQ Deadline-EQ.

(Closed) Open Item (155/86013-09): Deficiencies in EQ File of 3M Electrical Splice Tape-EQ.

(Closed) Open Item (155/86013-10): Deficiencies in EQ File of States Terminal Blocks-EQ.

(Closed) Open Item (155/86013-11): Licensee Walkdown of Limitorque Actuators in Response to IEN 83-72.

(Closed) Open Item (155/87006-01): NRC to Review Current Staffing Level for Site QA Organization.

(Closed) Open Item (155/87006-02): NRC to Review Audit Coverage of Commitments in 5.2.13 of N-18.7.

(Closed) Violation (155/88011-01): Violation of TS 9.0 Failure to Establish Adequate Program and Test Valves in Accordance with ASME Code Section XI.

(Closed) Open Item (155/88011-02): Review Licensee's Engineering Analysis and Resulting Long Term Corrective Actions.

3. Monthly Surveillance Observation (61726)

Station surveillance activities listed below were observed to verify that the activities were conducted in accordance with the Technical Specifications and surveillance procedures. The applicable procedures were reviewed for adequacy, test and process instrumentation was verified to be in their current cycle of calibration, personnel performing the tests appeared to be qualified, and test data was reviewed for accuracy and completeness. The NRC inspectors ascertained that any deficiencies identified were reviewed and resolved. The NRC inspectors observed the licensee's performance of the following surveillance tests on the indicated dates:

January 4, 1989,	T30-26 "Electric and Diesel Fire Pump L2 Module Test", Rev. 17, dated May 17, 1988.
January 26, 1988,	T30-03, "Monthly Drive Selector Valve Reduced Pressure Test", Rev. 10, dated May 18, 1988.
January 30, 1989,	T7-20, "Diesel Fire Pump Auto Start", Rev. 20, dated October 4, 1988.
January 30, 1989,	T7-21, "Standby Diesel Generator Start and Run Test", Rev. 16, dated March 16, 1988.
January 31, 1989,	T7-04, "Weekly Reactor Protection Logic System Test", Rev. 10, dated September 30, 1987.
January 31, 1989,	T7-18, "Bypass Valve Test", Rev. 10, dated May 18, 1988.
February 5,1989,	T7-03, "Control Rod Coupling Integrity Test at Power", Rev. 14, dated September 15, 1988.

No violations or deviations were identified in this area.

4. Monthly Maintenance Observation (62703)

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were certified; radiological and fire prevention controls were implemented.

The Senior Resident Inspector conducted several meetings with various maintenance personnel to discuss the "Notice of Proposed Rule Making on Maintenance of Nuclear Power Plants" and its impact on current plant maintenance operation. The discussion was well received and appreciated by the maintenance personnel.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority was assigned to safety related equipment maintenance which may affect system performance.

The NRC inspectors observed the licensee's performance of the following maintenance work orders on the indicated dates:

January 20, 1989, No. 89-ASD-0002, and accompanying jumper, link and by-pass (JLB) 89-001 and 89-002 "Battery switch connect to replace 70 AMP fuses with 100 AMP fuses."

January 21, 1989, No. 89-ASD-003 and JLB-89-004 "Main #1 SD disconnect switch replace the existing LPS - RK300 fuses with FRS-R-400 fuses using a temporary modification form."

January 21, 1989, No. 89-ASD-004 and JLB-89-003 "Panel 29 main disconnect replace the existing LPS-RK-150 fuses with FRS-R-200 fuses using a temporary modification form."

January 25, 1989, No. 88-SPS-0003, dated January 24, 1989, to investigate and repair circuit breaker OCB-1126 in the plant substation. The breaker had been running at a higher temperature than expected. Maintenance was completed on January 25, 1989.

January 27, 1989, No. 88-RCS-0014, dated March 9, 1988, to replace a section of the air line to the valve-operator on the discharge isolation valve (CV-4039) on the primary coolant system's cleanup pump.

January 30, 1989, No. 89-RMS-0001, dated January 16, 1989, for removing spent fuel-pool filter cartridges from the fuel pool and dumping them in the High Integrity Container (HIC) in the Waste Handling Building.

January 30, 1989, No. 89-CAS-0006, dated January 28, 1989, for investigating and repairing the No. 1 station air compressor, because of high aftercooler air temperature.

February 1, 1989, No. 88-RDS-0175, dated December 6, 1988, for testing and setting the steam drum level switches that sound the containment evacuation alarm.

February 2, 1989,	No. 89-NMS-0007, dated February 2, 1989, for calibrating the Wide Range Neutron Monitors to equal 100% power.
February 7, 1989,	No. 89-CLP-0009, dated February 5, for annual preventive maintenance on the reactor crane.

No violations or deviations were identified in this area.

5. Operational Safety Verification (71707)

The NRC inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during the inspection period. Instrumentation and recorder traces were examined for abnormalities and discussed with the control room operators, as were the status of control room annunciators. Reviews were conducted to confirm that the required leak rate calculations were performed and within Technical Specification limits. System Walkdowns were performed to verify the operability of the containment spray system. Tours of the containment sphere and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. Radiation protection controls were inspected, including Radiation Work Permits, calibration of radiation detectors, and proper posting and observance of radiation and/or contaminated areas. Security measures were inspected including access control of personnel and vehicles, proper display of identification badges for personnel within the protected area, and compensatory measures when security equipment had a failure or impairment.

The NRC inspector accompanied two separate Auxiliary Operators on their tours to observe them in the performance of their duties. The operators appeared to be knowledgeable and competent.

On January 13, 1989, the resident inspector was performing detailed inspection duties in the external cable penetration room. While inspecting the splices he noticed a small piece of insulation missing from an electrical splice exposing 1/8" of conductor on the containment water level indicator used in the Post-Incident/Emergency Core Cooling System. The licensee was promptly informed and took immediate corrective actions. Included in these actions was the entering of a 7 day Limiting Condition of Operation (LCO), repairing the splice, and investigation into the root cause. The licensee inspected 110 other splices in the cable penetration room and determined one other cable splice was marginal and performed the required repairs. The root cause was determined to be personnel error during installation in 1981. The licensee reviewed other work performed by the individual to ensure that no other problems existed with cable splices.

On January 20, 1989 an engineering analysis performed on the electrical systems associated with the Alternate Shutdown system (ASD) identified weaknesses in the circuit breaker coordination. A Corrective Action

Review Board (CARB) was held at 3:15 p.m. on 1/20/89 to review the Deviation Report. The Plant Review Committee met and approved the required fuse changes in the ASD system. It was determined at that time that the situation met the requirements of proposed Technical Specifications 11.3.5.3.5. The proposed Technical Specification states in part, "Station Battery System and Alternate Shutdown Battery System shall be operable under all conditions except during cold shutdown." This has a one hour LCO which could not be corrected in time.

At 4:15 p.m., the Shift Supervisor declared the Unusual Event and the Plant Manager immediately assumed the duties of the Site Emergency Director (SED). There was a partial activation of the Technical Support Center (TSC) with the On-Call Superintendent (acting as assistant SED), On-Call Technical Advisor, Dedicated Communicator and Technical Support Team. There was also a partial activation of Operation Support Center (OSC) with a Maintenance Supervisor and two repairmen. The Maintenance superintendent was also available.

Calls were made to the Charlevoix County Sheriff, the State Police, and the NRC by 4:23 p.m.

The Notification of Unusual Event was terminated at 6:30 p.m. when the correct size fuses were placed into service and the maintenance order closed.

The Senior Resident Inspector observed the replacement of the fuses and the return to operability of the ASD system. The licensee identified six more fuses that should be replaced. This put the licensee in a 72-hour LCO period. The SRI observed the replacement of these fuses at 7:00 a.m. on January 21, 1989. Full power operation was resumed later in the day.

On January 25, 1989, the licensee entered a three-day LCO for corrective maintenance because of high temperature readings on the breaker that normally powers station loads from the 138 KV line. The licensee's offsite crew was notified and arrived on site later in the day to perform the necessary repair. The repair was completed and the LCO exited later the same day.

No violations or deviations were identified in this area.

6. <u>Engineered Safety Feature System Walkdown (71710)</u>

The NRC inspectors verified the operability of the Liquid Poison System which is an Engineered Safety Feature system. The verification included a complete walkdown of the accessible portions of the system. Included were verification of valve labels, equipment condition, correct valve and breaker positions and apparent operability of support systems essential to the ESF system. A detailed review was conducted to confirm that the licensee's system lineup procedure matched the applicable as-built drawings; this included the following documents:

- Procedure 0-TGS-1-A-4, "Reactor Poison System Check Sheet," Revision 9, dated September 8, 1987.

- Drawing No. 0740G44004, "Liquid Poison System Valve Line-Up Diagram", Rev. 1, dated August 11, 1983.

- Drawing No. 0740G40107, "Piping & Instrument Diagram, Reactor Clean-Up, Shut-Down and Poison System", Rev. AH, dated October 27, 1988.

No Violations or Deviations were noted in this area.

7. Review of Metallurgical Report of Target Rock Valve Failure (73756)

Failure of a Target Rock relief valve on the Reactor Depressurization System (RDS) is documented in NRC Inspection Report No. 50-155/88011. Corrosion of the valve stem and guide is believed to be responsible for failure of the valve. The licensee provided several components from both the failed valve and others to Battelle Laboratories for chemical and metallurgical investigation.

Visual examination, cross sectional metallography, scanning electron microscopy, and micro-hardness methods were employed to determine the nature and extent of corrosion. It was found that the pitting attack was limited to the Stellite 6 hard facing on the stem and guide. The remainder of the materials, i.e., 304 stainless steel and carbon steel were generally unaffected. Microstructures of the materials were found to be typical of the alloys.

Microprobe analysis, electron, infrared, and spark source spectroscopy methods were utilized to determine the elements present in the pitted area and surface scale. X-ray diffraction and mapping techniques were used to determine the chemical form of major crystalline compounds and spatial distribution of elements respectively. Analyses of corrosion products identified primarily cobalt and mixed metal oxides. Some of the oxide present on the Stellite 6 was distinctly crystalline which may have caused the sticking of the valve.

In summary, the materials were found to be acceptable and the corrosion of the Stellite 6 is not unexpected in an oxygenated atmosphere. Commonly known corrosives such as sulfur and chlorine were not detected in any significant quantity.

8. Licensee Action On Information Notices (92701)

Information Notice No. 88-03 "Cracks in Shroud Support Access Hole Cover Welds," was closed during this inspection period based on a directive by the Division Director, Division of Reactor Safety, Region III. Our decision to close the item was based on the length of time the item has been in existence and the recognition of limited safety significance.

9. Temporary Instructions

a. (Closed) Temporary Instruction (TI) 2515/90 SIMS 41 "Inspection of Licensee's Implementation of Multiplant Action Item B-58, Scram Discharge Volume Capability." This inspection was performed to ensure scram discharge volume capability. The following paragraphs include inspector findings in regard to Temporary Instruction 2515/90, Inspection of Licensee's Implementation of Multiplant Action Item B-58, Scram Discharge Volume (SDV) Capability.

Most of the inspection requirements for this TI have been accomplished in prior inspections. These are documented in inspection reports 155/80009, 80010, 80015, and 80018.

The licensee responses to IE Bulletins 80-14 and 80-17 and their supplements addressed many of the items included in the Temporary Instructions.

Action 01. Determine if an engineering analysis or confirmatory letter was sent from the architect-engineer or vendor regarding the adequacy of size of SDV. An acceptable means of meeting this criterion is to provide a minimum scram discharge volume of 3.34 gallons per drive (GE letter OER 54 dated March 14, 1972). Also determine if the effects of hydraulic coupling were considered for those plants with the Instrument Volume (IV) located remotely from the SDV. IE Bulletin 80-17, Supplement 1, required this determination to be made. Credit may be taken for this review of Appendix B to the Generic SER that shows the status of this item for plants licensed before December 1980.

Licensee Response January 26, 1981.

The design of the Big Rock Point (BRP) scram discharge system is much different from the systems used in newer BWRs. This unique design (recognized in SER in Section 3.1.2 and 3.2) consists of a single, instrumented Scram Discharge Tank (SDT) rather than the separate scram discharge and instrument volumes. The SDT instruments are connected to an upper and lower two-inch instrument header that attaches directly to the SDT on the top and bottom through normally locked open manual isolation valves, not to a vent or drain line. The water discharged from the control rods during a scram is piped through four-inch branch headers (4) to a six-inch header and finally to the top of the SDT. Calculations were made to determine how many full scrams could be discharged into an empty SDT. These calculations were based on vendor instructions GEI-56217 because GE OER-54 (incorrectly stated as GE OER-52 in Attachment 1 - Design Criteria 1) is only applicable to BWR 2 through BWR 6 plants. It was concluded from the calculations that two full scrams of all control rods could be accommodated by an empty SDT. Therefore, no additional action was proposed with respect to this criteria.

Big Rock Dwg. No. 0740640122, Piping & Instrument Diagram Control Rod Drive System, shows the Dump Tanks having a 175 gallon capacity.

The effects of hydraulic coupling do not apply at Big Rock as explained in the Licensee Response of November 2, 1981 of design difference of Big Rock's scram volume.

Action 02. Review FSAR, Technical Specifications, and plant drawings to confirm that an automatic scram function exists for high instrument volume water level.

Licensee Response January 26, 1981

The amount of scram water discharge as described in Action 01 above allows the use of a SDT to receive the entire volume of water discharged during a scram. The Big Rock Point design uses a continuously monitored SDT which incorporates tank level annunciation and automatic scram (prior to inadequate remaining volume in the SDT to accept a full scram of all drives from the fully withdrawn position) instrumentation.

Big Rock Dwg. No. 074060122 also confirms the automatic scram function for high instrument volume water levels.

Action 03. Verify that safety-related Instrument Volume (IV) level instrument taps are on the IV only and not on connected piping above or below the IV. This may be confirmed by actual inspection or, if actual inspection is impractical, by review of current plant drawing.

The current plant drawing (Dwg. No. 0740640122) shows instrument taps on the IV and not on piping above and below the IV.

Action 04. Verify system configuration precludes a single line plugging or other single failure causing failure of the instruments to detect water in the IV. Determine if the IV level instrumentation and taps are redundant and determine if instrumentation connected to the taps is diverse. Diversity means that level is measured by sensors that employ different operating principles. Redundancy means that level is sensed by instruments using separate taps and independent power supplies to preclude a single failure from defeating their function.

Licensee Response November 2, 1981

To confirm that water accumulation in the dump tank does not significantly contribute to possibility of a failure to scram, a fault tree approach to evaluate the likelihood of accumulation, using probabilistic risk techniques was performed. Benefits obtained by installing diverse instrumentation regarding the risk associated with failure to scram was evaluated.

Section 5.3.8.3. of NUREG-0828, "Integrated Plant Safety Assessment Systematic Evaluation Program - Big Rock Point Plant," addressed the need for redundant and diverse level instrumentation on the scram discharge volume as identified by Generic Letter 81-18, dated March 30, 1981. As explained in Section 5.3.8.3., the Big Rock Point design uses four level switches on a single pair of 2-inch header pipes. Also, the scram discharge volume at Big Rock Point is actually a tank. The scram discharge volume design at other BWRs which Generic Letter 81-18 was intended to address is (1) a volume composed of piping with no sizeable tank, and (2) level switches on small instrument lines (typically quarter-inch lines). The Big Rock Point design is therefore quite different from the usual design and much less susceptible to the common mode failures addressed by the Generic Letter.

In conjunction with the above evaluation, NRR evaluated this issue using information from the Probabilistic Risk Assessment. Based on that evaluation, NRR concluded that modifications to provide redundant and diverse level instrumentation would not provide a cost effective improvement in the reliability of the present level instrumentation.

Therefore, NRR concluded that no modifications were required and that the issue of redundant, diverse scram discharge volume level instrumentation was resolved for Big Rock Point.

Action 05. Review licensee's analysis to determine whether the analysis considered water backup into the IV caused by the drain configuration or interface to other systems.

The drain from the dump tank dumps by free-fall into a larger drain. The dry well would have to be full of water prior to water backing up into the dump tank.

Action 06. Verify by review of current drawings or visual inspection that the IV vent and drain valves close on loss of air and that valve position is indicated in the control room.

Licensee response January 26, 1981

SDT vent and drain valve position indication is provided in the Control Room by a single set of lights for both valves. If the system shows a malfunction, the valve positions are physically checked (access to these valves is available during reactor operation). Drain valve closure may also be indicated by the SDT high-level alarm resulting from water accumulation in the SDT. Therefore, existing valve position indication in the Control Room is considered satisfactory. Drawing No. 0740640122 indicates that vent and drain valves close on loss of air.

Action 07. Verify by review of current drawings or visual inspection that an alarm exists in the control room for the presence of water in the IV. Verify that procedures exist for operator action in the event water is detected in the IVs.

Dwg. No. 074060122 indicates a control room alarm for the presence of water in the IV.

Licensee Response July 31, 1980

Some significant features of the Big Rock Point design should be understood when evaluating the results. The active SDV is totally within the Scram Dump Tank (SDT) having a 175 gallon capacity which is in excess of two full scram discharge quantities. The SDV piping is maintained water filled at all times due to a loop seal immediately ahead of SDT inlet. The SDT instrumentation provides a continuous monitor on water level. The system has a built-in safety feature of a high level alarm annunciator near the bottom of the tank (approximately 2" from the bottom) to ensure no appreciable water exists in the tank and as a backup to the "alarm", utilizes a high dump tank "scram" which places a trip into the Reactor Protection System when the tank is approximately one-half full (5/16" below centerline) thus unequivocally ensuring adequate margin for scram water is available. The high level scram feature precludes resetting the safety system in the "run" or "refuel" modes with an inappropriate amount of water in the tank.

Action 08. Verify by review of current drawings of visual inspection that a single active failure will not defeat isolation of the vent and drain valves. For example, check for redundant vent and drain valves.

Redundant vent and drain valves exist in the Big Rock design and these valves fail closed indicated in Dwg. No. 074060122.

Action 09. Verify that surveillance procedures exist that test operability of IV vent and drain valves. Periodic testing shall verify that valve closure time is less that 30 seconds (GE specification).

Licensee Surveillance Procedure TR-32 exists for testing the operability of IV vent and drain valves. Acceptance Criteria per this test is 20 seconds.

Action 10. Verify that procedures exist to have the level alarm and trip instrumentation tested in place. Check to confirm that steps in the procedures include demonstration of restoration of system configuration.

Licensee Response November 2, 1981

The present functional test performed at each refueling outage involves allowing the dump tank to fill following a scram signal during shutdown. This test results in filling the dump tank and verifies the communication path between the dump tank and instrument piping as well as operation of the level switches. No valve or instrument manipulation is required to perform this test and valve line-up is identical to that required for normal power operation.

Steps 6.2 and 6.3 are steps in procedure "Cleaning and Inspection of Scram Dump Tank High Level Scram Sensors", (IRPS-11), to ensure the system is restored to operable status.

Action 11. Verify that procedures exist to perform surveillance tests periodically and during each operating cycle as required by the plant Technical Specifications (TS). These tests should demonstrate scram instrument response and valve function at pressure and temperature at approximately 50% control rod density. (A suggested change to Technical Specifications acceptable to the NRC was transmitted by Generic Letter to All BWR Licensees from Darrel G. Eisenhut, Subject: Model Technical Specifications, July 7, 1980.)

Licensee Response January 26, 1981

SDT vent and drain valves operability and the SDT level detection instrumentation are tested at each refueling outage under the current surveillance test program. The evaluation identified a conflict between the surveillance test designations (each refueling) and the alternate test interval (12 months) specified by the Technical Specifications. The TS require that reactor safety system scram circuits (requiring plant shutdown to check) shall be conducted during each major refueling shutdown, but no less frequently than once every 12 months. This conflict has not resulted in a TS violation but was considered prudent to modify the surveillance test program to eliminate such an occurrence. The test status boards were changed to identify both the primary test interval (each refueling) and the alternate test interval (12 months).

b. (Update) (TI) 2515/73 "Inspection requirements for I.E. Bulletin 85-03, "Motor-Operated Valve Common Mode Failure during Plant Transients due to Improper Switch Settings.'" The purpose of the TI was to provide guidance for the inspectors' follow-up of the licensee's activities in responding to I.E. Bulletin 85-03. That bulletin, and Supplement 1, April 27, 1988, requested the licensees to develop and implement a program to ensure that switch settings on certain safety-related motor-operated valves are selected, set, and maintained correctly to accommodate the maximum differential pressures expected on these valves during both normal and abnormal events within the design basis. The licensee submitted the required interim report to the NRC by letter dated June 2, 1988. This interim report summarized the licensee's program and schedule for complying with Bulletin 85-03 and Supplement 1. The report was reviewed and accepted by the NRR in a letter sent to Region III dated January 18, 1989.

The licensee completed the corrective actions identified in the interim report and submitted a final report to the NRC on January 20, 1989. The resident inspectors reviewed the procedural changes to verify the completed corrective actions. This bulletin, and its supplement, will remain open until the licensee's final report is reviewed and accepted by NRR.

c. (Closed) TI 2500/26 SIMS BL-87-02 "Inspection Requirements for NRC Compliance Bulletin 87-02, Fastener Testing to Determine Conformance with Applicable Material Specifications". The purpose of the TI was to ensure fasteners selected in response to NRC Bulletin 87-02 were representative of installed fasteners and that suspect fasteners were selected before testing. The licensee's responses to Bulletin 87-02 were completed and documented in a letter to the NRC dated January 12, 1988. The licensee's responses to Supplements 1 and 2 of Bulletin 87-02 were completed and documented in a letter to the NRC dated July 14, 1988. The Temporary Instruction required the following actions.

(1) Review the licensees receipt inspection program/procedures for safety related and non-safety related fasteners to determine what characteristics are inspected and compare it to the licensee's description provided in the bulletin's response.

(2) Review the licensee's maintenance/warehouse procedures for issue and control of safety related and non-safety related fasteners and compare it to the licensee's descriptions provided in the bulletin response.

(3) Participate in the licensee's selections of samples as required by paragraph two of Bulletin 87-02.

(4) Review the licensee's description of further action being taken as required by actions in paragraph two of Bulletin 87-02.

Action items C and D are documented in Inspection Report 155/88002. The Senior Resident Inspector performed action items A and B and determined that the licensee has in place the appropriate procedures for procurement, receipt, storage, and issuance of fasteners. The program is as described (with minor improvements) in the licensee's responses to NRC Bulletin 87-02, including Supplement 1 and 2 and was adequate to insure that the correct fasteners are issued and used in the intended application. This completes the requirements of TI 2500/26 and NRC Compliance Bulletin 87-02. Both items are closed.

d. (Closed) TI 2515/100 "Proper Receipt, Storage, and Handling of the Emergency Diesel Generator (EDG) Fuel Oil." The purpose of the TI was to inspect the licensee's program on EDG fuel oil stored on site to ensure that the licensee was maintaining an adequate quality.

To verify that the licensee has a program in place to purchase and store fuel oil that meets TS requirements the inspection verified the following:

(1) That the licensee routinely determines the quality of stored fuel oil with effective scheduled analyses.

(2) That the licensee can detect degradation of stored fuel oil quality, as may be indicated by excessive water accumulation, oxidation, or biological contamination, among other possible causes of degradation.

(3) That the licensee routinely monitors and cleans filters, strainers, and other components prone to fouling in the fuel oil system.

(4) That the licensee routinely reviews and evaluates NRC information on this subject.

The Senior Resident Inspector completed the TI 2515/100 Appendix A questionnaire which listed the above requirements to verify that the licensee has a program in place to purchase and store fuel oil that meets the TS requirements. The licensee's program incorporates standards ASTM D975 and ANSI N195 but not Regulatory Guide 1.137. The Senior Resident Inspector reviewed purchase order requirements and surveillance procedures T90-10, Diesel Generator and Fire Pump, Diesel Fuel Storage Tanks Sampling and Analysis, TR-84 Emergency Diesel Generator Inspection and Repair, and TR-85 Diesel Fire Pump Inspection and Repair to verify that the procedures were sufficiently detailed for the licensee's personnel to perform the activities. These were documented in the Appendix A questionnaire, which was sent to NRR on February 10, 1989. This item is closed.

10. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) throughout the month and at the conclusion of the inspection period and summarized the scope and findings of the inspection activities. The licensee acknowledged these findings. The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents or processes as proprietary.