U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-155/86014(DRP)

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

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, MI 49201

Facility Name: Big Rock Point Nuclear Plant

Inspection At: Charlevoix, MI 49720

Inspection Conducted: September 24 - November 26, 1986

Inspectors: S. Guthrie
M. Parker

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

12-11-86
Date

Inspection Summary

Inspection on September 24 - November 26, 1986 (Report No. 50-155/86014(DRP))
Areas Inspected: Routine, unannounced inspection conducted by the Senior Resident Inspector of Licensee Actions on Previous Inspection Findings, Operational Safety, Maintenance, Surveillance , Training, IE Bulletins, Licensee Event Reports, Licensing Actions, and Regional Requests.
Results: Of the ten areas inspected, no violations or deviations were identified. One significant safety concern identified and is discussed in Section 3.f.

DETAILS

1. Persons Contacted

*D. Hoffman, Plant Superintendent
G. Petitjean, Planning and Administrative Services Superintendent
*G. Withrow, Engineering Maintenance Superintendent
*R. Alexander, Technical Engineer
*R. Abel, Production and Plant Performance Superintendent
*L. Monshor, Quality Assurance Superintendent
R. Barnhart, Senior Quality Assurance Administrator
P. Donnelly, Senior Review Supervisor, Nuclear Activities Department
D. Swem, Senior Engineer
G. Sonnenberg, Shift Supervisor
D. Staton, Shift Supervisor
W. Trubilowicz, Operations Supervisor
*J. Beer, Chemistry/Health Physics Superintendent
E. Evans, Senior Engineer
R. Brady, Senior Plant Technical Analyst
J. Tilton, General Engineer
D. Kelly, Maintenance Supervisor
D. Ball, Maintenance Supervisor
W. Blosh, Maintenance Engineer
M. Acker, Senior Engineer
J. Kneeland, Reactor Engineer
L. Darrah, Shift Supervisor
J. Horan, Shift Supervisor
R. May, Shift Supervisor
R. Scheels, Shift Supervisor
J. Warner, Property Protection Supervisor
T. Fisher, Senior Quality Assurance Administrator
S. Bartosik, General Quality Assurance Consultant
R. Krchmar, General Quality Assurance Analyst
*E. Raciborski, Planning and Scheduling Administrator

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

*Denotes those present at exit interview.

2. Licensee Action on Previous Inspection Findings

A review of long standing open or unresolved items (including but not limited to items which are greater than three years old) was made to determine if there were any items for which the expenditure of additional inspection effort was not justified. On the basis of this review, the following items are administratively closed out:

155/840xx-03 - GE Part 21 report on Loctite
155/81009-01 - Systems description manual not current
155/83014-01 - Long term action on control rod blades

155/84007-01 - Procedural inadequacy on rad waste alarms
155/82012-02 - Facial hair policy for fire brigade members
155/82012-04 - Qualification of fire watches for offsite personnel
155/82012-05 - Review of MO's for transient fire loading
155/82012-06 - SCBA's referenced in implementing procedures

No violations or deviations were identified in this area.

3. Operational Safety Verification

The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the inspection period. The inspector verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. 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. The inspector by observation and direct interview verified that the physical security plan was being implemented in accordance with the station security plan.

The inspector observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection period, the inspector walked down the accessible portions of the Liquid Poison, Emergency Condenser, Reactor Depressurization, Post Incident, Core Spray and Containment Spray systems to verify operability.

a. On September 21 the reactor reached the all rods fully out configuration and power coastdown commenced. Power coastdown will continue until commencement of refueling on January 2, 1987. The 1987 refueling was delayed by approximately two months to allow for delivery of nuclear instrumentation components required for a scheduled replacement of source range nuclear instrumentation. The inspector verified the recalibration of power range instrumentation to reflect the declining maximum power representing 100% power as registered on the picoammeters.

b. During a regular review of licensee deviation reports the inspector learned of a contaminated carpet in the access control building front lobby on September 24. The carpet, which was contaminated to 1000-2000 cpm in several spots, was discovered after an upright vacuum being brought to access control was frisked and found to have 300 cpm above background on the base of the bag. Root cause of the incident was determined by the Corrective Action Review Board to be inadequate or improper frisking. The licensee's immediate corrective action included removal and controlled storage of the carpets and vacuum bag and survey of carpets in other plant areas. Proposed actions to prevent recurrence include revision of contamination survey schedule to provide for weekly survey of carpeted offices areas and the control room and a survey of footwear worn by plant employees only within the protected area with disposal of those found to be contaminated. The Chem/HP Superintendent issued a memorandum to all employees describing the incident. The licensee is investigating the availability of improved models of hand and foot frisking equipment. At the close of the inspection period approximately two-thirds of the employee footwear had been examined for contamination and procedural changes increasing the survey frequency of carpeted areas were in routing for management concurrence.

c. On September 25 the inspector observed portions of unannounced drug testing activities conducted on site.

d. On September 30 the inspector observed activities to transfer spent demineralized resins from their storage tank to the concentrate storage tank in order to make room in the resin discharge tank for future discharges. Both tanks are underground and accessible by manholes. Appropriate radiological controls were evident during the transfer operations, including coverage by a Radiological Protection Technician.

e. During the inspection period the inspector observed the arrival of components for the self contained filtration system for the spent fuel pool. The system uses a floating saucer-like skimmer with adjustable buoyancy to provide suction from the fuel pool surface for a system of pumps and filters. Except for the floating surface unit, all components are underwater in the spent fuel pool. An area to mount liners for radioactive waste shipping containers will be prepared underwater in the fuel pool, permitting all filter changing and storage to be conducted underwater. The system is scheduled to be installed and tested prior to the upcoming refueling outage and is expected to have a significant impact on the licensee's man-rem reduction program.

f. During the week of October 27 the Inspector reviewed the licensee's assessment of leaking fuel bundles presently installed in the reactor core. Fuel leaking in the "H" series fuel has become a regular occurrence in recent operating cycles, and the licensee has in each instance conducted "fuel sipping" analyses of suspect fuel bundles to locate and replace defective fuel rods. With the present cycle drawing to a close and cesium isotopic analysis of primary coolant indicating leakage of an estimated two or three fuel rods in older "H" series fuel bundles, the licensee has elected to forego sipping activities and reinstall 38 "H" series bundles. The licensee recognizes that some fuel bundles with leaking fuel rods may be reinstalled in the core, creating the likelihood of further leakage in the next cycle. In addition to the possible reappearance of old leaks from previous core exposure, the licensee speculates that additional leakage will be observed.

Fuel leakage is evidenced by increased rates of off gas released to atmosphere via the plant stack. Typical values for non-leaking fuel are in the 200-300 uCi/sec range, while off gas release currently runs at approximately 3000 uCi/sec. The air ejector off-gas monitor activates a control room alarm at 30,000 uCI/sec and isolates the off gas shutoff valve at 50,000 uCi/sec, a set point sufficiently low to ensure the dose at and beyond the site boundary does not exceed the annual dose limits imposed by 10 CFR 20 to unrestricted areas. The licensee estimates that one year of operation at the 30,000 uCi/sec value would approach but not exceed the limit of 10CFR 50, Appendix I. Big Rock Off-Normal Procedure ONP 2.17, Abnormal Off Gas or Stack Gas Release, requires load reduction when off gas activity reaches 40,000 uCi/sec and the initiation of an orderly shutdown upon reaching a release rate of 50,000 uCi/sec.

Big Rock has exhibited a history of fuel leakage over the last three operating cycles. In 1984 fuel leaks related to defects in cladding of foreign manufacture resulted in release rates of approximately 27,000 uCi/sec and led to voluntary reactor power restrictions. All fuel with the suspect clad material was removed from the reactor. (Reference Reports No. 155/84005, 155/84007, 155/85002 and LER 155/84002.) Beginning in November, 1984, and continuing through two operating cycles, the unit experienced fuel leakage not related to material defects but attributed to pellet-clad interaction resulting from changes in core physics design. Prior to the approval of new core physics models by NRR, the reactor had been operated at approximately 210-215 MWt in order to provide a conservative margin against exceeding thermal limits. Following approval, the reactor was operated in the 230-232 MWt range. Thus, "H" series fuel which had delivered satisfactory performance at the lower power levels has apparently not withstood the increased thermal megawatt output. This "H" series fuel rod design deficiency has been corrected in the "I" series fuel, which decreased the pellet-to-clad clearance and increased helium overpressure to three atmospheres to enhance heat transfer characteristics. "I" series fuel is not suspect in the current fuel failure investigation.

The Big Rock Core consists of 84 fuel bundles. The next core reload is compromised of:

- 40 "I" series bundles, including 20 new unexposed bundles
- 4 rebuilt "H" bundles
- 2 "G" bundles of an older design similar to "H" but not suspected of fuel rod leakage
- 38 "H" bundles of the type exhibiting leakages, some undetermined number of which are known to leak.

The licensee based the decision not to perform sipping operations to identify leaking fuel bundles on the following considerations:

(1) The operational characteristics of "H" series fuel, including leakage mechanisms, are well established after several cycles of sipping for leaking bundles. Those failure characteristics typically involve minor leakage from a single fuel rod within the bundle and historically results in an offgas activity

increase of approximately 1000 uCi/sec per single fuel rod. The conclusion is that projected leakage would not differ substantially from previous leakage and would not cause activity levels approaching those requiring administrative action. The licensee discounts the probability of a bundle with minor fuel rod leakage reinstalled in the core failing catastrophically during the next cycle.

(2) Licensee calculations indicate that the total gaseous activity released to the atmosphere if leakage rate change continues at its present rate and severity would yield a dose to persons in the site area lower in magnitude than the dose obtained by Operators performing sipping operations. Specifically, the licensee estimates the person-rem exposure to a limited number of persons performing fuel inspection and sipping operations to be five times the person-rem exposure to the general population during the next operating cycle.

(3) Licensee calculations indicate a high probability that older "H" bundles considered most likely to be leaking are being retired from the fuel cycle. These calculations are based on cesium isotopic analysis.

The inspector expressed the following concerns to the licensee:

The licensee's decision relies heavily on an assumption that the type, severity, and frequency of fuel rod failures will remain unchanged during the upcoming operating cycle. Although two cycles worth of data gleaned from sipping operations and isotopic analysis provide some confidence in predicting the type of failure, the assumption that frequency of leakage among the 38 "H" bundles will remain constant may or may not be valid. Continued exposure in the core may, in fact, accelerate the rate of leakage or result in more significant failures.

While recognizing the ALARA significance of the ten person-rem exposure estimated to inspect and sip the entire 84 bundle core, only 38 "H" bundles are suspect. Since isotopic analysis has confidently identified leaking bundles as being of the "H" vintage, sipping only those 38 "H" bundles scheduled for reinstallation would logically yield the greatest amount of information for the least personnel exposure. If licensee predictions that all currently leaking fuel will be discharged at the end of the present operating cycle is accurate, total exposure estimates from sipping the 38 "H" bundles would be further reduced.

While recognizing that minor fuel failures are not uncommon throughout the nuclear industry and that past releases have been well below administrative and regulatory limits, the uncertainties associated with a core reload with some degree of fuel failure is not representative of a conservative approach to reactor operation.

The inspector consulted with Region III and NRR fuels experts. During a conference call involving those groups and the licensee on November 25 the staff of NRR related their concern that water leaking inside failed fuel rods would lead to corrosion and increase the likelihood of major failures which release fission products to the coolant. Release of fission products under similar circumstances in other reactor plants has resulted in considerable exposure to personnel from the turbine and steam piping. The staff noted the general understanding with plants throughout the United States is that no fuel known to have failures will be reinstalled in the core.

In a second conference call involving the same parties, the licensee reiterated their position that ALARA considerations make the decision not to inspect for leaking fuel the preferred choice. The licensee committed to again review their decision making process in an attempt to identify a means of providing the staff the reasonable assurance it seeks that leaking fuel will not be returned to the core.

g. On October 29 the licensee informed the inspector of fixed contamination areas identified in a warehouse area being refurbished for material storage after several years use as an office area. Prior to use as office space the area had been used for general and contaminated storage, and several small areas of fixed contamination of up to 20,000 counts/minute contact were painted over and then covered with carpeting. The highest calculated dose from the most contaminated spot in the area formerly occupied as office space was 160 mr/hr(Beta). Other contaminated spots were located in areas not previously occupied as office space. Each area of fixed contamination is being removed by chipping away the surface of the concrete floor. The licensee noted that personnel assigned to the office area wore personnel dosimetry at all times and that based on the exposure history of the individuals, no increasing trends indicating radiation levels greater than background could be identified. The area was appropriately posted for fixed contamination upon discovery.

h. On November 14 the inspector observed the licensee's annual exercise conducted in conjunction with two local hospitals with which the licensee has letters of agreement to provide treatment for persons injured on site. The drill, which is required by Section 9.8.1.2.E of The Site Emergency Plan, was preceded by training conducted November 13 for drill participants, including members of local ambulance crews on a voluntary basis. The drill scenario simulated a contaminated individual dressed in anti-contamination clothing suffering chest and arm injuries from a fall. Communication between the control room and access control was effective to provide the prompt entry of the ambulance into the protected area. The exercise involved transportation of the simulated patient to the local hospital and simulated treatment by Emergency Room medical professionals wearing anti-contamination clothing. Proper security control over the ambulance and attendants while within the protected area was observed, and radiological controls to minimize the spread of contamination was in evidence at the site, in the ambulance and at the hospital. The drill scenario was developed and observed on site and at the hospital by the medical consultants under contract with the licensee.

i. On November 24 the inspector observed licensee response to indications of a minor steam leak that was later determined to be a packing leak on valve VP-13, the reactor vessel isolation valve for the three inch liquid poison line to the reactor. During daily leak rate calculations the licensee received indication of a step increase in leak rate that caused an increase from the expected .3-.4 gpm result to a rate of 0.812 gpm. Administrative procedures require power reduction and investigation at leak rates in excess of 0.8 gpm. Reactor power was reduced from the all rods out maximum of approximately 53 MWe to approximately 3 MWe, equivelant to station loads. Additional four hour leak rate calculations were commenced while investigation was underway. The cause of the leak was identified and, when attempts to tighten the packing did not improve packing integrity, the valve was placed on its backseat, resulting in immediate control of the leak. The result of the second leak rate calculation was 0.924 gpm, slightly below the 1.0 gpm limit for operation established by Technical Specifications. Additional post maintenance leak rate calculations indicated the rate had returned to normal values. The relatively small release of airborne activity was far below regulatory limits.

   Subsequent to completion of repairs and return to maximum power a licensee review of leak rate calculation data hour by hour indicated that for a period of approximately two hours the leak rate may have exceed the 1.0 gpm limit up to a maximum of 1.17 gpm. No unusual event declaration based on exceeding a Technical Specification limit was made because discovery of the calculated high leak rate was made well after repairs had been successfully completed. The licensee did notify NRC Headquarters of their findings. A review by the inspector concluded that failure to identify the high leak rate during the relatively small time frame in which it was above the limit does not necessarily reflect a deficiency in the way leak rates were calculated. In this instance a conscientous Shift Supervisor was able to reconstruct and review plant parameter data over an extended time period spanning the event to identify a trend.

j. The inspectors reviewed the licensee's program for cold weather preparations to ensure the licensee has maintained effective implementation of protective measures for extreme cold weather committed to in response to IE Bulletin 79-24.

   The inspectors performed a review and walkdown on systems susceptible to freezing to verify the presence of heat tracing, space heaters, and/or insulation; the proper setting of thermostats; and that heat tracing and spaceheating circuits had been energized. The inspectors reviewed systems subject to maintenance and/or modifications during the last year to verify that protective measures had been established.

The inspectors reviewed the licensee's response to IE Bulletin 79-24, frozen lines, dated October 31, 1979, and were satisfied that the licensee is taking adequate protective measures as described in the response.

Review of the licensee's cold weather check-off list indicated that all required action had been completed.

A walkdown of systems susceptible to freezing identified that the screen house heaters were not in service and that the breaker had been tagged out for ground fault. Also one of the two vent shed heaters was out of service for repairs. The inspectors observed that at the close of the period the licensee was taking the necessary steps to return this equipment to service. The inspectors also identified to the licensee that the check-off list had not been modified to require the necessary checks of the alternate shutdown building, although the heating system was in service.

No violations or deviations were identified in this area.

4. Monthly Maintenance Observation

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 properly certified; radiological controls were implemented; and, fire prevention controls were implemented.

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

a. During the period the inspector reviewed licensee preparations for the 1987 refueling outage due to commence January 1. Preplanning took into consideration the Maintenance Order Back Log impact on outage preparation. Planners established a monthly schedule of pre-outage maintenance activities for the period September - December. The schedule included tests and inspections of cranes, lifting gear, and equipment used for fuel handling, preparation of tools and work areas and winterization of buildings and equipment. Spare pumps and control rod drives required to be rebuilt were identified and scheduled. During the week of November 4 the licensee conducted spent fuel pool cleanup activities. The cleanup was in preparation for refueling activities, the removal and shipment of old fuel channels previously crushed and prepared, and the installation of the new self-contained fuel pool filtration system.

b. During the period the inspector observed repairs to the plant heating boiler, now used to provide heat for office areas but in past years used to provide steam used in reactor plant startup. Minor residual internal contamination from the boiler's past role in reactor operation required radiological controls, and appropriate coverage was observed. Electrical noise from boiler welding activities resulted in a short period alarm signal in the control room which originated in channel six, one of two start-up range channels which measure power from source level to 10^{-6}% of reactor power. Channel seven was not affected. The short period scram is a function of the intermediate range instrumentation and is bypassed at higher power levels above the startup range. At the full power operation the trip signal has no effect. Technical Specification 6.1.5(h) permits normal shutdown with only one start-up range operable. The licensee has not undertaken to repair problems with channel six because the entire source range nuclear instrumentation, including the suspect cables in channel six, is scheduled for replacement during the January 1987, refueling.

c. During the inspection period the inspector monitored licensee efforts to assess and trend pump condition based on vibration analysis. The licensee's program uses measurements conducted and interpreted quarterly. Based on trends throughout 1986 the licensee determined the need to overhaul the motor on Reactor Feed Pump "A", which will be performed during the outage. In addition, the need was identified to rebuild the "A" Service Water Pump based on increasing vibration. Knowledge of component condition for the eleven pumps and engines monitored by the program was factored into maintenance decisions and represents a significant step in addressing preventive maintenance programmatic inadequacies first presented in Report No. 155/85002(DRP) in which the licensee's preventive maintenance activities were considered to be reactive rather than preventive in nature.

d. On September 24 the inspector observed portions of the repairs to terminal blocks on MO-7080, backup firemain to Core Spray Heat Exchanger. Terminal block replacement and rerouting of wires to relieve tight radius bends were performed after the deficiencies were identified during an inspection for Environmental Qualifications of Electrical Equipment (EQ) the preceeding week. The inspector reviewed with the licensee acceptable approaches to obtaining EQ certification for future spare parts purchasing including:

(1) Obtain certification of qualification from the vendor.

(2) Subject samples of the material to actual test conditions that duplicate the harsh environment predicted for the worst case accident to which the component might be subjected.

(3) Perform a thorough analysis that shows the material matches the exact composition of material in a comparable component which is documented as being qualified.

For procurement of items which meet a Military Specification (MILSPEC), a formal process of documentation must be performed by knowledgeable engineering personnel to verify that the environmental conditions referenced in the MILSPEC meet or exceed those which the component would be subjected to in the postulated worst case accident.

e. On September 24 the inspector observed portion of the overhaul of house service air compressor No. 2.

f. On October 21 the licensee was unable to successfully complete the routine weekly Reactor Protection System (RPS) logic test. During performance of the test, control room alarms not anticipated or associated with the test were received and would not readily reset. Investigation revealed that the No. 1 RPS motor-generator set voltage regulator was delivering low voltage to power RPS circuitry. Voltage regulator repairs were completed and the RPS logic test successfully completed.

g. During the period October 28 - November 4 the inspector observed construction of a security barrier at a location within the protected area. Compensatory measures to ensure vital area integrity were evident until construction activities were completed. Barrier adequacy will be inspected in future security inspections.

h. On September 29 the inspector observed activities to reinforce the Lake Michigan Shoreline with stone to retard erosion caused by wave action and high lake levels.

During the period the inspector reviewed the licensee's program to investigate the impact of high lake levels on the screen house and equipment contained therein. Presently the reactor is to be shut down in accordance with Emergency Operating Procedure EMP 3.9, Flooding, if uncontrollable water accumulation threatens the availability of the electric or diesel fire pump or if leakage in the circulating water pumps expansion joints is observed. Site Emergency Implementing Procedure No. 1, Classification of Emergency Conditions, requires declaration of an alert should water level exceed 583.5 ft. (screenhouse floor level). During operation of both circulating water pumps the screenhouse intake bay level is approximately three feet below lake water elevation. Recent surveys have established current lake level at 582.6 ft., less than one foot below the level at which an alert declaration is required. Licensee research predicted a 1987 monthly mean lake level of 583.3 ft. assuming normal regional precipitation and meteorological trends, and cautions that flucuations of one foot or more above the monthly mean are possible in any given day.

The licensee's study identified equipment likely to be in jeopardy and recommended alternatives to avoid emergency actions in the short term and ensure equipment integrity for the duration of the plant's operating license. Proposed alternatives emphasized rebuilding the foundation to raise the height of electrical cabinets, fire pump batteries, canal sample pump, motor control centers, fire jockey pump, and demineralized water pump. At the close of the period the licensee was evaluating available alternatives.

i. On November 3 the inspector observed maintenance activities to correct erratic behavior of the channel one picoammeter that caused an erroneous high flux alarm and scram signal. Big Rock Reactor System design requires high flux signals from two of the three power range channels to generate a scram signal that results in control rod insertion, and failure of channel one resulted in a half scram signal. Investigation determined the failure was caused by broken wire in the nuclear instrumentation feedback circuitry. A similar problem with channel two feedback circuitry resulted in a reactor scram on July 2, 1986. (Reference Report No. 155/86007, Section 6). Repairs were completed and operability testing of channel one was verified prior to return to service. The licensee had earlier intended to complete a facility modification that would replace older nuclear instrumentation, including the problem power range feedback circuitry, during the January, 1987, outage. That project has been scaled back to replacement of source range circuitry only due to material availability problems, with the power range portion of the project deferred until 1988. Inspection and repair of power range instrumentation as required will be included in 1987 outage activities.

j. On November 6 the inspector observed portions of maintenance activities associated with removal of contaminated filter socks from the spent fuel pool filter system. The work was accomplished with appropriate radiological coverage and concern for personnel exposure.

k. During the week of November 17 the inspector observed licensee's activities to locate and repair a small leak in off-gas monitor piping. That piping normally operates under a vacuum, but operators on November 16 observed an increase in turbine area airborne activity when the off-gas piping was subjected to its daily purge with instrument air. Activity levels during purging indicated an increase on the turbine area continuous air monitor of approximately 1000 counts/minute representing primarily short lived nuclides. To find the small leak an enclosure of lead bricks was constructed to shield a detector from background activity while technicians pumped air from the piping area through the detector. Appropriate radiological controls were in evidence for persons involved in the maintenance activity. However, radiation protection personnel at the scene resisted the inspector's initiative to restrict access to the turbine area during the purging period, maintaining that the activity levels were far below maximum permissible concentrations and the area was already posted as a radiation area. The inspector expressed a concern that an individual in the area who was unaware of the increased airborne activity would likely be unnecessarily exposed, an approach not in keeping with the ALARA concept. Radiation protection personnel pointed out that they had not advised the Operations Department to impose area restrictions or broadcast warnings for previous purgings after the leak was discovered and would not restrict work activities in this instance. Prior to commencement of the purge, Control Room Operators, on their own initiative, announced the purge and warned personnel to avoid the area, a conservative measure similar to warnings regularly broadcast when radioactive sources used for detection calibration are moved through a normally occupied area. Leak detection activities eventually identified the three way valve as leaking during the purge cycle. Valve repair eliminated the problem.

l. On November 21 the inspector reviewed with the licensee modifications to the Reactor Depressurization System (RDS) pilot valves and top assemblies, scheduled for the 1987 outage. Problems associated with corrosion from carbon steel components lodging under seating surfaces in the pilot valve have resulted in valve leakage in past operating cycles and made quarterly operability testing of system valves difficult. Working closely with the valve manufacturer the licensee intends to replace existing pilot valves and carbon steel top assemblies with stainless steel top assemblies featuring pilot valves with a specification change that doubles the force exerted by the spring holding the pilot disc into the main disc. Modification of the pilot disc guide sleeve is expected to reduce cavitation caused by leaking steam, thereby reducing pilot seat erosion. A stronger coil is provided to ensure the capability of compressing the heavier spring. The licensee has ordered four valves and intends to use the eight existing top assemblies as spares. Production acceptance testing performed by the manufacturer shows satisfactory pilot valve operation using 60 vdc to operate against 1700 psi steam pressure. Normal plant voltage for pilot valve operation is 120 vdc and normal plant pressure is 1335 psi, indicating a margin of confidence that that valve will operate as designed with the heavier spring. The modification will not require change in bench testing procedures for leakage of top assemblies prior to installation. Replacement of all top assembly bolts and nuts is planned for later in 1987. In addition the licensee is evaluating the need for replacement or refurbishment of discs and seats in the main portion of the valve and devising a means to perform full stroke valve testing of the valve while in place in the system. Present testing procedures for the main valve result in instantaneous lifting of the main disc off its seat using a charge of high pressure air.

No violations or deviations were identified in this this area.

5. Licensee Event Reports

By letter dated October 3, the licensee submitted Licensee Event Report (LER) 86006 describing procedural inadequacies resulting in untimely initiation of a fire watch patrol required by Technical Specifications 12.3.7.12. The requirement that a fire watch patrol be established within one hour to perform hourly inspection when one or more fire barriers protecting safety related areas are not functional was not met. During performance of Surveillance Test TR-69 (Fire System Inspection performed during refueling outages), on October 6-7, 1985, Shift Supervisors (SS) did not recognize that minor deficiencies identified during the surveillance constituted a non-functional fire barrier. Review of the deficiencies by the Fire Protection personnel on October 8, 1985, resulted in determination that the barrier was indeed not functional and in immediate establishment of the fire watch patrol. The delay was identified through a Quality Assurance audit and presented to plant management August 25, 1986. Reportability of the event under the 30-day requirement of 10 CFR 50.73 commenced on that date. An extension to the 30 day requirement was granted by Regional Management on September 24, 1986. All necessary repairs had been completed by the August 25th determination of reportability.

The licensee concluded that the surveillance procedure was deficient in that (1) it failed to provide the SS with a clear definition of what constituted a non-functional fire barrier and (2) that no instructions were provided concerning the need for prompt action when deficiencies were identified. The licensee's action to prevent recurrence included (1) addition of a precaution to TR-69 to require immediate notification to the SS of defects or holes in fire barriers, and (2) a note in the body of the procedure identifying the need to establish a fire watch patrol within one hour of the discovery of any defect in a fire barrier. These procedure changes are to be implemented prior to the next performance of TR-69. In accordance with 10 CFR Part 2, Appendix C, Section V.A, a notice of violation will not be issued for this violation because it meets all of the following tests:

a. It was identified by the licensee. (By a Quality Assurance Audit)

b. It fits in Severity IV or V. (This would have been a V)

c. It was reported, if required. (By LER 86006)

d. It was or will be corrected, including measures to prevent recurrence, within a reasonable time. (A fire watch was established and the surveillance procedure was changed)

e. It was not a violation that could reasonably be expected to have been prevented by the licensee's corrective action for a previous violation. (No such violation could be identified)

6. Surveillance

On November 14 the inspector observed the night time performance of Surveillance T7-21, weekly Start Test of the Standby Diesel Generator. The standby diesel generator is located outside the protected area and has no automatic start features. The normal means of performing the surveillance was modified at the request of NRR to require the licensee to demonstrate the ability to start the generator without the use of installed area illumination, using only hand held flashlights. Demonstrated ability to start the generator using only flashlights for illumination is a factor in a pending NRR decision to exempt the licensee from the Appendix R requirement for installed emergency area lighting for the generator building and normal path of travel between the control room and generator location. The issue is presented in Report No. 155/85022(DRS), section 7.b.2. The licensee successfully demonstrated the ability to start and operate the standby diesel generator using only a hand held flashlight.

7. Training

a. At a series of Department Safety Meetings, the licensee conducted training in Radiation Work Permit (RWP) use and appropriate personal contamination monitoring (frisking) practices. The subject of RWP training was to explain recent changes in RWP use that moves away from RWP as a dose tracking devise and toward use of the RWP as a tool for informing workers what radiological considerations are important to accomplish assigned radiological work.

b. On September 24 the licensee conducted training for twenty-five persons assigned under the Site Emergency Plan as communicators and dose assessors. The personnel were trained by State Police officials in methods to effectively communicate with State officials during plant emergencies. Supplementing the presentation the Big Rock Emergency Preparedness Coordinator conducted walkthroughs with communicators and those with clerical assignments to provide hands-on experience with forms and telephone communications.

c. By letter dated September 24 the Big Rock Point Training Department received notification that the accreditation board of the Institute of Nuclear Power Operation (INPO) had fully accredited five employee training programs. The accreditation included designation of the department as a branch of the National Academy for Nuclear Training. The approved program include those used to train licensed reactor operators, senior reactor operators, and non-licensed operators, instrument and control technicians, and on-call technical advisors. The accreditation process was the result of a two year review by INPO that included evaluation of the plant's program against INPO objectives and criteria, on site INPO observation of training activities and facilities, and observation of simulator training for licensed operators and technical advisors.

8. Licensing Activities

a. By letter dated October 28 the staff of NRR approved the licensee's request to postpone by approximately three months the replacement of Squib valve primer and trigger assemblies in the Liquid Poison System. Technical Specification 5.2.3 requires periodic test firings and restricts primer and trigger assemblies to a maximum of five years service, a requirement based on the vendor recommendations for replacement five years after date of manufacture. The licensee has consistently met the scheduled testing and replacement requirements. They anticipated questions regarding the life cycle of the currently installed primer and trigger assemblies in light of the three month delay in commencement of the next refueling outage. The NRR decision to permit the extended service period was based on the vendor's written approval of the 3-4 month service life extension and documentation by the licensee that the assemblies had never been exposed to adverse environmental conditions.

b. By letter dated October 28 the staff of NRR determined the licensee's proposed Intergranular Stress Corrosion Cracking in Stainless Steel piping (IGSCC) weld inspection sample size to be acceptable. The licensee is required by Generic Letter 84-11 to inspect or reinspect 20% of subject welds, or a minimum of four previously uninspected and two previously inspected welds for each pipe size. The licensee's original submittal of 18 proposed weld inspections was found to be unacceptable, and their revised approved submittal commits to the examination of 31 welds during the 1987 outage, a figure compatible with the requirements of Generic Letter 84-11. The licensee last conducted IGSCC inspections during the September-October, 1985, outage and of the 18 welds tested in recirculation and shutdown cooling systems piping no defects were identified. (Reference Report No. 155/85018).

c. By letter dated November 17 the staff of NRR informed the licensee that Topic III-1 of the Systematic Evaluation Program for Big Rock Point dealing with classification of structures, systems, and components, had been resolved. The staff's review of licensee submittals concluded that an adequate evaluation has been performed to demonstrate sufficient margins of safety exists in the quality standards used for design and construction of the facility.

9. Regional Request

a. During the period the inspector reviewed, at the request of Region III, licensee response to IE Bulletin 86002, Static "O" Ring (SOR) Differential Pressure Switches to determine if the licensee's submittal addressed applications important to safety or only safety related applications. The Regional request resulted from a review by IE Headquarters of responses from several licensees which determined that many licensees incorrectly interpreted the subject of the Bulletin as only SOR differential pressure switches which were used in safety related applications. The intent of the bulletin was to review the application of SOR switches installed as electrical equipment important to safety, as defined by 10 CFR 50.49(b). The Big Rock Point response addressed the Bulletin as written by reviewing all applications of SOR Series 102 or 103 differential pressure switches installed in systems subject to Technical Specifications, and concluded that no switches of the models specified were installed in application subject to Technical Specifications. The licensee's response is reviewed in Section 6 of Report No. 155/86011.

The inspector's review determined that the licensee's original submittal addressed all applications of the subject model switch. The inspector reviewed the plant equipment list and substantiated the licensee's findings that SOR switches of the specified model are installed in two locations in the turbine bypass valve control system. One switch (DPS-7911) monitors filter differential pressure and the other (DPS-7912) monitors accumulator pressures. The sole function of each switch is to provide control room alarms. The inspector concluded that the licensee's original submittal adequately addressed all applications of the subject switch.

b. At the request of Region III the inspector, on November 12, reviewed the licensee's program for ensuring the quality of splices using heat shrinkable tubing in order to determine the extent of splice deficiencies. The licensee has been using Raychem splices since 1980 and, until April, 1986, used the vendor's application instructions as a guide to appropriate installation. In April the vendor's application guide was incorporated into a plant specification that includes drawings for different splice configurations and a check sheet that preplans each splice and requires engineering department approval before the work is begun. Drawings and instructions for tubing overlap, shims, and color coding of spliced wires is attached to the maintenance procedure that accompanies the workmen in the field. The licensee has three vendor qualified instructors on site. The licensee stated that they are not aware of any splicing deficiencies at the facility. The inspector concluded that problems with heat shrinkable tubing in evidence at other facilities do not appear to exist at Big Rock Point.

c. At the request of Region III and Headquarters the inspector reviewed the licensee's response to IE Bulletin (IEB) 86-01, Minimum Flow Logic Problems That Could Disable RHR Pumps. The inspector's initial review of IEB 86-01, which supports the licensee's conclusion that the concerns raised by the Bulletin are not applicable to Big Rock because of a significant design differences between Big Rock and newer plants with an RHR system, is presented in section 7.b of Report No. 155/86007.

As requested, the inspector verified the licensee submitted their initial response within seven days of the receipt of the bulletin. Also, since the licensee's conclusion that the bulletin is not applicable is based on the absence of an RHR system, the inspector reviewed with the licensee the potential for pump damage among those ECCS pumps available at Big Rock Point that would result from prolonged pump operation with no discharge path or recirculation line. At Big Rock four pumps contribute to core spray and containment spray capability: one diesel fire pump (DFP), one electric fire pump (EFP), and two post-incident (PI) core spray pumps. On the DFP and EFP, relief valve capacity installed at the discharge of each pump provides protection against pump damage at shut off head. The PI pumps are not equipped with relief capacity or minimum flow lines. A flow path exists to provide some recirculation flow through the core spray test tank, but that flow path is normally isolated from the normal flow path by locked closed valves and only tested during refueling outages. The design feature that provides for no automatic starts of the PI pumps means that all PI pump and valve operations are manual at the component location or remote manual from the control room, indicating a margin of confidence that PI pumps would not run for extended periods against a closed discharge path. System Operating Procedure (SOP)-8, Post Incident System, specifically addresses all valves in the discharge path to establish a flow path except for Core Spray Valve MO-7061. Because the PI portion of core spray capability would not be expected to be placed into operation until well into the accident sequence at a point where sufficient water had accumulated in the bottom of the sphere, Valve MO-7061 would have already been positioned. The inspector concluded that the absence of minimum flow lines on pumps used for core spray purposes did not appear to degrade the plant's ability to inject water into the reactor.

d. At the request of Region III the inspector reviewed the licensee's response to Information Notice 86-72, Failure of Stainless Steel Springs in Valcor Valves Due to Hydrogen Embrittlement. The notice grew out of a 10 CFR 21 report from the vendor and did not require a licensee response. The licensee's review indicated that the Notice was not applicable to Big Rock Point. The inspector verified with the licensee that a search of the computerized equipment database indicated no Valcor valves were installed anywhere at Big Rock Point.

e. At the request of Region III the inspector reviewed the licensee's response to General Electric Service Information Letter (SIL) 445, Intermediate Range Monitor (IRM) Fuse Failure. The SIL describes an event at another boiling water reactor in which IRM channels became inoperable after fuses on both the positive and negative dc power supplies blew on a voltage surge. For a brief period operators at that facility were unaware of IRM inoperability because replacement of fuses on the positive side only cleared control room alarms. In its cover letter transmitting the SIL to Big Rock Point, General Electric informed the licensee that the information was not applicable to Big Rock Point and was being forwarded for information purposes only. Big Rock Point has indicating lights and volt meters on the IRM high voltage power supplies.

f. The inspector reviewed the proposed Facility Change to Source Range Nuclear Instrumentation scheduled for completion during the 1987 outage to determine if any portion of the new SIL was applicable. The new power supplies have indication of 120 V power into the unit and loss of that power supply generates a control room alarm. This configuration is considered adequate to allow operators to accurately determine the status of source range nuclear instrumentation. The existing IRM and Power range power supplies will be replaced with a wide range instrumentation designed to span both ranges during the 1988 refueling outage.

No violations or deviations were identified in this area.

12. Exit Interview

The inspector 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 inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents or processes as proprietary.