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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 2, 1980

Docket No. 50-155

Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Hoffman:

Enclosed is the staff's evaluation of the implementation of "Category A" Lessons Learned requirements (excluding 2.1.7a) at the Big Rock Point Plant. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on February 28 and 29, 1980.

Based on our evaluation, we conclude that, with the exception of Lessons Learned Items 2.1.6.B, 2.1.8.A, and 2.2.2.B, the implementation of the "Category A" requirements at Big Rock Point Plant is acceptable. The acceptability of your plans regarding implementation of these three items will be addressed in response to your letters of February 22, 1980 and April 2, 1980, regarding the Overall Risk Assessment. Certain items, identified in the evaluation, will be verified by the Office of Inspection and Enforcement.

This evaluation does not address the Technical Specifications necessary to ensure the limiting conditions for operation and the long-term operability surveillance requirements for the systems modified during the "Category A" review. You should be considering the proposal of such Technical Specifications. We will be discussing this item with you in the near future.

Sincerely,

Dennis M. Crutchfield for

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Safety Evaluation

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POOR ORIGINAL

EVALUATION OF LICENSEE'S COMPLIANCE WITH CATEGORY "A" ITEMS OF NRC RECOMMENDATIONS RESULTING FROM TMI-2 LESSONS LEARNED

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONSUMERS POWER COMPANY
BIG ROCK POINT NUCLEAR PLANT

DOCKET NO. 50-155

Date: April 15, 1980

I. INTRODUCTION

By letter dated October 17[1] and 30[2], November 21[3] and December 27[4] 1979 and January 18[5] and February 22[6], and March 14 [7], and April 2, 1980[8], the Consumers Power Company (licensee) submitted commitments and documentation of actions taken at Big Rock Point Plant to implement our TMI-2 Lessons Learned (L^2) requirements which are discussed in NUREG-0578. To expedite our review of the licensee's actions, members of the staff visited Big Rock Point on February 28 and 29, 1980. This report is an evaluation of the licensee's efforts to implement each Category "A" requirement which was to be completed by January 1980.

Implementation of our short term requirements is complete with the exception of three items which the licensee has requested to be deferred pending completion of on overall risk assessment[6]. The NRC is currently evaluating this request. These items are addressed in our evaluation. Another item which the licensee has requested to be deferred is the installation of a containment hydrogen monitor which is a long term Category B requirement.

II. EVALUATION

Each of the Category "A" requirements is identified below. The staff's requirements are set forth in Reference 9; the acceptance criteria is given in Reference 10. The numbered designation of each item below is consistent with the identifications in NUREG-0578.

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS (Power-Operated Relief Valves and Reactor Water Level Instrumentation)

The NRC requirement, as it is applicable to BWR's, is that provisions must be made such that the power operated relief valves can be supplied emergency power when offsite power is not available. Further, for air-operated valves, emergency power must be available to the air compressors in order to provide a long term supply of air. The reactor water level instrumentation must also be capable of operating from emergency power.

The Big Rock Point design includes pilot-operated relief valves in the reactor depressurization system. The pilot valves are electrically operated. Each pilot-operated relief valve is in series with a normally closed, air-operated gate valve. Each pilot-operated relief valve is powered from vital power supply.

The air-operated gate valves are supplied from the plant air system through double check valves. Redundant air compressors for the air supply can be manually connected to vital buses. The gate valves, however, fail open on loss of air.

The reactor water level instrumentation for safety system activation and control is powered from the vital buses.

Based on our review of the above design, we conclude that the licensee meets our emergency power supply requirements for relief valves, air operated valves and reactor water level instrumentation.

2.1.2 PERFORMANCE TESTING FOR RELIEF AND SAFETY VALVES

Originally, the licensee committed to a test program being conducted by the GE BWR Owners Group for performance testing of BWR relief and safety valves. By letter dated March 14, 1980, the licensee informed us that the ongoing testing program being conducted by the Electric Power Research Institute (EPRI), appears most appropriate for Big Rock Point and it is pursuing this matter with EPRI to determine whether Big Rock Point can be enveloped by the EPRI test program. The results of either test program will cover Big Rock Point relief and safety valves. At present both these programs are under review to ensure that the NUREG-0578 requirements are met.

Based on our review of the above information, we believe that our requirements for performance testing of relief and safety valves will be satisfied. Completion of these test programs is on a schedule different from Category "A" items. Therefore, we conclude that the licensee has met the Category "A" requirements of this item.

2.1.3.a DIRECT INDICATION OF POWER OPERATED RELIEF VALVE AND SAFETY VALVE POSITION

The pilot-operated relief valves and the air-operated gate valves in the reactor depressurization system have stem mounted qualified switches which provide direct position indication of these valves in the control room. To meet our requirement for safety valves the licensee has installed an acoustical monitoring system to monitor the position of each of the six safety valves. This acoustical monitoring system is similar to those found acceptable by the staff for this purpose for other power reactors. Each valve is monitored by a single accelerometer. A charge amplifier amplifies each accelerometer output. The signal is then sent to a signal processing unit mounted in the control room. Valve position indication and annunciation for each monitored valve is available on the main control room panel. The licensee has stated that the valve position indication components will be seismically and environmentally qualified by October 1980 for conditions applicable to Big Rock Point.

The backup valve indication is provided by a common drain header high temperature alarm and containment high-pressure alarm. The licensee has stated that these indirect methods of recognizing an open valve are discussed in Plant Operating Procedures.

Based on our review of the licensee's design, we conclude that the licensee has met our requirements for this item. Our Office of Inspection and Enforcement will verify (1) the adequacy of installation of the above design, (2) the adequacy of the qualification documentation of the valve position indication components and, (3) the adequacy of the procedures for backup valve position indication. This will be documented in an appropriate inspection report.

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING FOR PWR'S AND BWR'S

The NRC requirements, licensee actions and our evaluation thereof for this item are reported separately by the NRC Bulletins and Orders Task Force in NUREG-0626 which is incorporated herein by reference.[11]

2.1.4 CONTAINMENT ISOLATION

The NRC lessons learned requirements concerning containment isolation direct the licensee to: a) determine which systems penetrating containment are considered essential or non-essential to safety; b) modify containment isolation circuitry to automatically isolate all non-essential systems by diverse parameters; and c) modify containment isolation circuitry to assure that clearing of the containment isolation signals does not cause inadvertent opening of containment isolation valves. In addition, the isolation system was reviewed to assure that certain systems which are isolated but might be desir to use following an accident or transient, can be reopened and to assure that operator controls of containment isolation are not ganged to reopen multiple systems with a single operator action.

The licensee has identified the systems penetrating containment which are considered to be essential as follows: Post Accident and Fire Water Supply System; Post-Accident Back Up System; Ventilating Vacuum Breaker Sensing Line; and Core Spray Recirculation System. These systems are the basic accident mitigation systems at Big Rock Point. Those systems that could be used to provide water to the reactor, such as the Reactor Feedwater and the Control Rod Hydraulic Systems are not classified as essential by the licensee for purposes of this review. The licensee states that classification of systems as essential and unessential and the modifications required to eliminate the procedural control of the isolation valves will be reviewed as a part of NRC Systematic Evaluation Program.

All systems with automatic isolation valves isolate on diverse signals. Either low reactor vessel water level or high containment pressure results in isolation. Typically each penetration is served by one automatic isolation valve and one remote manual valve in series.

Certain systems with liquid flow directed into containment, such as Reactor Feedwater and the Control Rod Hydraulic Systems are isolated with two check valves in series to prevent back flow and would be operable if it were desirable to continue water inflow following a containment isolation. The demineralized water system has a check valve and a remote manual air operated valve in series to provide isolation. The containment building heating system and the service water system are closed systems, and have no isolation provisions. The service air and instrument air supplies have a check valve in series with a local manual valve.

The containment isolation system at Big Rock Point has been modified by the installation of reset circuits on four systems (nine valves) to prevent inadvertent automatic opening of the containment isolation valves following clearing of the containment isolation signals. With the modification, it is necessary for the operator to actuate a reset button to allow operation of the valve. The reset button has no effect if the containment isolation signal is present. If the containment isolation signal has cleared, actuation of the reset button puts the isolation valve in the position selected by the valve control switch. Procedures call for the valve control switches of containment isolation valves to be placed in the closed position following an automatic containment isolation. With this administrative control, at least two independent operator actions are required to reopen any valve. Each reset button affects the operation of a single system.

At Big Rock Point, no systems were identified which after isolation would be necessary to reopen following an accident.

We conclude that the modified containment isolation system meets our requirements. Verification of the adequacy of the design modifications and the modified containment isolation procedures will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.5.a DEDICATED H_2 CONTROL PENETRATIONS

Big Rock Point is not required by license to have recombiners or containment purging for post-accident combustible gas control of the containment atmosphere. Therefore, this item does apply to this plant.

2.1.5.b INERTING BWR CONTAINMENTS

This item does not apply to Big Rock Point.

2.1.5.c H_2 PURGE PROCEDURES

Big Rock Point does not use recombiners or containment purge for post-accident combustible gas control of the containment atmosphere; therefore, this item does not apply to this plant.

2.1.6.a SYSTEM INTEGRITY

The post-incident core spray system is the only plant system outside containment that would or could contain highly radioactive fluids during a serious transient or accident. The applicable parts of this system that are outside containment have been recently tested by the licensee. The licensee reported that there was no observable leakage. This ESF system will be tested during each refueling outage.

The licensee has reviewed the plant for potential leakage release paths from the above system and other systems. This is for leakage due to design and operator deficiencies as discussed in the NRR letter to the licensee regarding North Anna and Related Incidents dated October 17, 1979. No corrective actions to the plant were needed.

Based on the above considerations, we conclude that the licensee has met our requirements for this item. There are no Category "B" requirements. Verification of the adequacy of the procedures for the periodic leakage testing of the core spray system will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.6.b PLANT SHIELDING REVIEW

The licensee has completed a radiation and shielding review of the spaces around the Big Rock Point containment and the only plant system outside containment, the post-incident core spray system, which would or could contain highly radioactive fluids in a serious transient or accident. Maps of the estimated dose rates onsite and in different areas of the plant outside the containment have been provided. The radioactive source terms assumed in the review are those given in the NRC letter to the licensee dated October 30, 1979.

During the initial hours after an accident movement around the site may be precluded because of radiation from the thin steel containment vessel. The licensee's calculations show that initial radiation levels in most plant areas may be greater than 10^3 R/hour. The licensee has reviewed his plant to determine what plant modifications are needed to limit personnel radiation exposure in vital areas to less than 10 CFR Part 20 levels. The proposed modifications being considered are: (1) installing local shielding for specific plant areas, (2) erecting a concrete shield building around the containment or (3) installing local shielding and a shield building. By letters dated February 22(7) and April 2 (8), 1980, the licensee has proposed to defer further action on plant shielding for the site until completion of an overall plant risk assessment program. The NRC is currently reviewing this proposal. After completion of this review, the aspects of this proposal as it is related to the requirements of NUREG-0578 will be set forth. Therefore, our conclusions on this Category A requirement of this item will be deferred until the completion of the NRC review.

Irrespective of the above , the licensee has completed his review of vital areas in which personnel occupancy may be limited by radiation during post-accident operations. The control room, the interim Technical Support Center and the Operational Support Center are sufficiently shielded that they would remain accessible for continuous occupancy. The vital areas in which personnel occupancy may be limited are the backup emergency diesel, backup cooling water supply hose to the core spray heat exchanger and the emergency diesel general fuel supply. Plant modifications will be made to enable the backup systems to be placed inservice from the control room and to provide a larger fuel supply tank for the emergency diesel. These modifications are Category "B" requirements which should be completed by January 1981.

The licensee has not completed his review of radiation qualification of safety equipment which may be unduly degraded by radiation during post-accident operations. The licensee has provided a tabulation of this equipment and the radiation dose to which each piece of equipment may be subjected to during an accident. The post-accident radiation exposure of some ESF equipment outside containment may depend on site shielding discussed above. The equipment which would not be affected by this shielding has been identified by the licensee and the review will be completed by May 15, 1980 for electrical equipment and July 15, 1980 for mechanical equipment. Plant modifications to protect this safety equipment are Category "B" requirements and should be completed by January 1980.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements of this item with the exception of the items deferred to the risk assessment proposal. An evaluation of the licensee's design review and corrective actions will be performed as part of the review of the Category "B" requirements for this item.

2.1.8.a POST-ACCIDENT SAMPLING

The licensee has performed a design and operational review of reactor coolant and containment atmosphere sampling systems. The licensee does not have the capability of obtaining either sample outside containment. At this time, the licensee does not have an adequately shielded area, where a post-accident sampling station could be located.

The licensee has developed interim sampling methods to quantify the radioactivity released from the core. These interim methods are the following: (1) obtain a sample from the core spray recirculation lines outside containment (i.e. for less than 10% core damage) and (2) obtain an estimate of the direct radiation through the thin-wall containment sphere by a radiation monitor with a readout in the control room. The licensee has procedures to collect samples during post-accidents conditions. The above monitor for the containment is in place and procedures have been written for its use.

The licensee has performed a design and operational review of the plant radiological analyses and chemical analysis facilities under post-accident conditions. These facilities may not be accessible as discussed in Item 2.1.6.b and there are no areas onsite that equipment could be moved to do an analysis. The licensee does have procedures to analyse post-accident samples if the facilities are accessible.

By letters dated February 22[7] and April 2[8], 1980, the licensee has proposed to defer the design and installation of a shielded post-accident sampling facility outside containment until completion of an overall plant risk assessment program. The NRC is currently reviewing this proposal. After completion of this review, the aspects of this proposal as it is related to the requirements of NUREG-0578 will be set forth. Therefore, our conclusions on this Category A requirement of this item will be deferred until the completion of the NRC review.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements for this item with the exception of the items deferred to the risk assessment proposal. Verification of the adequacy of the interim procedures will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.8.b HIGH RANGE EFFLUENT MONITORS

The licensee has an interim method to quantify high level noble gas effluent from the plant stack during an accident. The plant stack is the only post-accident effluent release point that can be monitored. The interim monitor is located adjacent to the current stack gas sample and return lines. The monitor is shielded and approximately 13 feet below grade to eliminate post-accident background radiation. The radiation readings from the detector are continuously displayed in the Operations Support Center and can be communicated to the control room by telephone. The licensee has procedures to convert the monitor readings to the rate of noble gas radioactivity being released from the stack. These procedures provide conversion values as a function of time after core shutdown to account for the decay of the short-lived noble gases.

The current stack effluent monitoring location may not be accessible during post-accident conditions to collect radioiodine/particulate sample cartridges. This may continue for several days into an accident. The lack of access to plant areas following an accident with a significant loss of radioactivity to the containment is discussed in Item 2.1.6.b. The licensee has procedures which address the removal of the radioiodine/particulate cartridges when they are accessible during post-accident conditions.

The licensee has an interim method to quantify high level radioiodine and particulate effluents from the plant during an accident. This is for the case when stack effluent radioiodine/particulate cartridge can not be collected. The licensee has written procedures to estimate releases from the plant by sampling airborne radioactivity in accessible areas around the plant.

The licensee has committed to providing an automated stack effluent sampling and analysis system for post-accident conditions.

Plant modifications to provide the capability to quantify post-accident radioiodine/particulate is a Category "B" requirement which should be completed by January 1981.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements for this item. Verification of the adequacy of the procedures and equipment installation to quantify high-level post-accident effluents will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.8.c IMPROVED IN-PLANT RADIOIODINE INSTRUMENTATION

The licensee has a standard gross monitor with silver zeolite cartridges which is in the control room/Technical Support Area to promptly analyze air samples for radioiodine during an accident. The licensee will provide by 1981 portable gamma spectrum analyzers to improve this radioiodine monitoring capability.

Based on this, we conclude that the licensee has met our requirements for this item. There are no Category "B" requirements. Verification that the licensee has the above equipment in place and is periodically checked and calibrated and the plant staff is trained in the use of these monitors will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

3.1.a SHIFT SUPERVISOR RESPONSIBILITIES

The licensee has issued a management directive which states that the shift supervisor has primary management responsibility for the plant and is the only person authorized to direct licensed activities and licensed operators. He is required to remain in the control room at all times during accident situations until properly relieved. Procedures have been revised such that duties which detract from the primary responsibility for safe operation of the plant have been assumed by another individual.

Emergency planning procedures recognize four different situations including: unusual event, alert, site emergency and general emergency. The licensee would allow the Shift Supervisor to leave the control room for a brief period of time under the less serious conditions of unusual event and alert, should the Shift Supervisor determine that his absence was in the best interests of plant safety. We find this acceptable.

We conclude that the licensee has met our requirements for this item. Verification of the licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented by an appropriate inspection report.

2.2.1.b SHIFT TECHNICAL ADVISOR

For the interim period of 1980, the licensee has provided an on-shift technical advisor (STA) to assist the shift supervisor in the function of accident assessment. The STA will also perform the operational assessment function. All current STAs are engineers with bachelor's degree or equivalent and have received seven weeks of training related to the accident assessment function, including one week of simulator training. The STA's will be kept informed of current plant status by duties which include formal shift turnovers, watch relief logs, and other plant condition logs. They will work eight hour shifts and will be on site at all times during this duty to be available in the control room within 10 minutes of being called.

For the long term, the STA position will continue to be filled by engineers with bachelor's degrees or equivalent who will be expected to commit to two years in this position. A comprehensive training program has been outlined.

Based on the above considerations, we conclude that the licensee has met the requirements of this item. Verification of the adequacy of the implemented procedures will be performed by the Office of Inspection and Enforcement and will be documented by an appropriate inspection report.

2.2.1.c SHIFT AND RELIEF TURNOVER PROCEDURES

The licensee has revised plant procedures to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the on-coming shift.

Based on the above considerations, we conclude that the licensee has met the requirements of this item. Verification of the adequacy of the implemented procedures will be performed by the Office of Inspection and Enforcement and will be documented by an appropriate inspection report.

2.2.2.a CONTROL ROOM ACCESS

The licensee has implemented procedures which will limit control room access during an emergency. The shift supervisor is responsible for maintaining control of personnel entering the control room. He is authorized to refuse entry or direct personnel to leave the control room. During an accident the shift supervisor remains in the control room at all times to direct the activities of the control room operators. He may be relieved by another qualified shift supervisor. The shift supervisor will limit the control room access to only those personnel who are essential for the direct operation of the plant and to those required to support plant operation during the emergency conditions.

On the basis of our review, we conclude that the licensee has satisfied our requirements of this item. The Office of Inspection and Enforcement will verify the adequacy of the implemented procedures. This will be documented in an appropriate inspection report.

2.2.2.b ON-SITE TECHNICAL SUPPORT CENTER (TSC)

The licensee has established an interim onsite technical support center located in the area of the shift supervisor and assistant plant superintendent's office. This area is just outside the control room. Direct communication between TSC and the control room, and the NRC have been established. Portable instruments are provided within the TSC and/or control room to monitor both direct radiation and airborne radioactive containments. Since the location of the TSC is adjacent to the control room, plant parameters necessary for accident assessment can be directly observed from the control console instruments through the control room large viewing windows. The TSC also has access to piping and instrumentation diagrams. The licensee has also discussed its plans to have a permanent TSC, however, by letters dated February 22(7) and April 2(8), 1980, the licensee has proposed to defer further action until completion of an overall plant risk assessment program. The NRC is currently reviewing this proposal. After completion of this review, the aspects of this proposal as it is related to the requirements of NUREG-0578 will be set forth. Therefore, our conclusions on this Category A requirement of this item will be deferred until the completion of the NRC review.

Based on our review of the licensee's submittal and our site visit, we have concluded that the licensee met our requirements for this item with the exception of the item deferred to the risk assessment proposal. The Office of Inspection and Enforcement will verify that the adequacy of the procedures for activation of the TSC and directing the operation of the TSC. This will be documented in an appropriate inspection report.

2.2.2.c OPERATIONAL SUPPORT CENTER (OSC)

The air compressor room has been designated as the Operational Support Center. This space is separate from the control room and is shielded in a manner similar to the control room. Communication to the control room is via the plant telephone system. The licensee has revised its emergency plan to reflect use of this area as OSC.

Based on our review of the licensee's submittal and our site visit, we conclude that the licensee has met our requirements for this item. The Office of Inspection and Enforcement will verify the adequacy of the licensees revised procedures to include this center and its use in the emergency plan. This will be documented in an appropriate inspection report.

NRR ITEM: REACTOR COOLANT SYSTEM HIGH POINT VENTS

The licensee has committed to install remotely operated reactor coolant system high point vents. The proposed design provides venting by installation of remotely operated vent valves on each of the emergency condenser tube bundles. Each vent path will be controlled by two solenoid operated valves in series. Both valves in a given vent path are controlled by a single switch and powered from a single power supply. The valves fail closed on loss of power. Single failure criteria for operation is provided by the redundant vent path with a separate power supply and control switch. Single failure criteria for isolation is provided by the emergency condenser steam and condensate return valves. Indication of power applied to open the vent valve is available in the control room. Plant modifications are Category B requirements which should be completed by January 1981.

Based on the above considerations, we conclude that the licensee has met the requirements of this item.

REFERENCES

1. Letter, CPCo (Bixel) to NRC (Z/NRR), dated October 17, 1979.
2. Letter, CPCo (Bixel) to NRC (Z/NRR), dated October 30, 1979.
3. Letter, CPCo (Hoffman) to NRC (Z/NRR), dated November 21, 1979.
4. Letter, CPCo (Hoffman) to NRC (Z/NRR), dated December 27, 1979.
5. Letter, CPCo (Fields) to NRC (Z/NRR), dated January 18, 1980.
6. Letter, CPCo (DeWitt) to NRC (D/NRR), dated February 22, 1980.
7. Letter, CPCo (Hoffman) to NRC (Z/NRR), dated March 14, 1980.
8. Letter, CPCo (DeWitt) to NRC (D/NRR), dated April 2, 1980.
9. Letter, NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, dated September 13, 1979.
10. Letter, NRC (Denton) to ALL OPERATING NUCLEAR POWER PLANTS, dated October 30, 1979.
11. NUREG-0626, Report of the Bulletins and Orders Task Force.