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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated March 26, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 158 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective prior to restart from the next refueling outage but no later than August 31, 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

for L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 20, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 158

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| VII | VII |
| XIII | XIII |
| 3/4 3-10 | 3/4 3-10 |
| 3/4 3-11 | 3/4 3-11 |
| 3/4 3-27 | 3/4 3-27 |
| 3/4 3-28 | 3/4 3-28 |
| 3/4 3-29 | 3/4 3-29 |
| 3/4 3-30 | 3/4 3-30 |
| 3/4 5-9 | 3/4 5-9 |
| 3/4 5-10 | 3/4 5-10 |
| 5-5 | 5-5 |
| B 2-5 | B 2-5 |
| B 3/4 3-1 | B 3/4 3-1 |
| B 3/4 3-1a | B 3/4 3-2 |
| B 3/4 3-1b | B 3/4 3-3 |
| B 3/4 3-1c | B 3/4 3-4 |
| B 3/4 3-2 | B 3/4 3-5 |
| B 3/4 3-3 | B 3/4 3-6 |
| B 3/4 3-4 | B 3/4 3-7 |
| B 3/4 3-5 | ----- |
| B 3/4 5-2 | B 3/4 5-2 |
| B 3/4 5-3 | B 3/4 5-3 |

## INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## INDEX

BASES

SECTION PAGE

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| 1. Manual Reactor Trip | NOT APPLICABLE |
| 2. Power Range, Neutron Flux (High and Low Setpoint) | Less than or equal to 0.5 seconds* |
| 3. Power Range, Neutron Flux, High Positive Rate | NOT APPLICABLE |
| 4. Power Range, Neutron Flux, High Negative Rate | Less than or equal to 0.5 seconds* |
| 5. Intermediate Range, Neutron Flux | NOT APPLICABLE |
| 6. Source Range, Neutron Flux | NOT APPLICABLE |
| 7. Overtemperature delta T | Less than or equal to 6.0 seconds* |
| 8. Overpower delta T | Less than or equal to 6.0 seconds* |
| 9. Pressurizer Pressure--Low | Less than or equal to 1.0 seconds |
| 10. Pressurizer Pressure--High | Less than or equal to 1.0 seconds |
| 11. Pressurizer Water Level--High | Less than or equal to 2.0 seconds |

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| 12. Loss of Flow - Single Loop (Above P-8) | Less than or equal to 1.0 seconds |
| 13. Loss of Flow - Two Loops (Above P-7 and below P-8) | Less than or equal to 1.0 seconds |
| 14. Steam Generator Water Level--Low-Low | Less than or equal to 1.5 seconds |
| 15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | NOT APPLICABLE |
| 16. Undervoltage-Reactor Coolant Pumps | Less than or equal to 1.2 seconds |
| 17. Underfrequency-Reactor Coolant Pumps | Less than or equal to 0.6 seconds |
| 18. Turbine Trip | |
| A. Low Fluid Oil Pressure | NOT APPLICABLE |
| B. Turbine Stop Valve | NOT APPLICABLE |
| 19. Safety Injection Input from ESF | NOT APPLICABLE |
| 20. Reactor Coolant Pump Breaker Position Trip | NOT APPLICABLE |

## TABLE 3.3-5

## ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | | RESPONSE TIME IN SECONDS |
|---|---|---|
| 1. Manual | | |
| a. | Safety Injection (ECCS) | Not Applicable |
| | Feedwater Isolation | Not Applicable |
| | Reactor Trip (SI) | Not Applicable |
| | Containment Isolation-Phase "A" | Not Applicable |
| | Containment Purge and Exhaust Isolation | Not Applicable |
| | Auxiliary Feedwater Pumps | Not Applicable |
| | Essential Service Water System | Not Applicable |
| b. | Containment Spray | Not Applicable |
| | Containment Isolation-Phase "B" | Not Applicable |
| | Containment Purge and Exhaust Isolation | Not Applicable |
| | Containment Air Recirculation Fan | Not Applicable |
| c. | Containment Isolation-Phase "A" | Not Applicable |
| | Containment Purge and Exhaust Isolation | Not Applicable |
| d. | Steam Line Isolation | Not Applicable |
| 2. Containment Pressure-High | | |
| a. | Safety Injection (ECCS) | Less than or equal to 27.0@@/27.0++ |
| b. | Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. | Feedwater Isolation | Less than or equal to 8.0 |
| d. | Containment Isolation-Phase "A" | Less than or equal to 18.0#/28.0## |
| e. | Containment Purge and Exhaust Isolation | Not Applicable |
| f. | Auxiliary Feedwater Pumps | Not Applicable |
| g. | Essential Service Water System | Less than or equal to 13.0#/48.0## |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| **3. Pressurizer Pressure-Low** | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0@@/27.0++ |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation-Phase "A" | Less than or equal to 18.0# |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Auxiliary Feedwater Pumps | Not Applicable |
| g. Essential Service Water System | Less than or equal to 48.0++/13.0# |
| **4. Differential Pressure Between Steam Lines-High** | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0@@/37.0@ |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation-Phase "A" | Less than or equal to 18.0#/28.0## |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Auxiliary Feedwater Pumps | Not Applicable |
| g. Essential Service Water System | Less than or equal to 13.0#/48.0## |
| **5. Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low** | |
| a. Safety Injection (ECCS) | Less than or equal to 29.0@@/39.0@ |
| b. Reactor Trip (from SI) | Less than or equal to 5.0 |
| c. Feedwater Isolation | Less than or equal to 10.0 |
| d. Containment Isolation-Phase "A" | Less than or equal to 20.0#/30.0## |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Auxiliary Feedwater Pumps | Not Applicable |
| g. Essential Service Water System | Less than or equal to 15.0#/50.0## |
| h. Steam Line Isolation | Less than or equal to 13.0 |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 6. Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0@@/37.0@ |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation-Phase "A" | Less than or equal to 18.0#/28.0## |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Auxiliary Feedwater Pumps | Not Applicable |
| g. Essential Service Water System | Less than or equal to 14.0#/48.0## |
| h. Steam Line Isolation | Less than or equal to 11.0 |
| 7. Containment Pressure--High-High | |
| a. Containment Spray | Less than or equal to 45.0 |
| b. Containment Isolation-Phase "B" | Not Applicable |
| c. Steam Line Isolation | Less than or equal to 10.0 |
| d. Containment Air Recirculation Fan | Less than or equal to 660.0 |
| 8. Steam Generator Water Level--High-High | |
| a. Turbine Trip | Less than or equal to 2.5 |
| b. Feedwater Isolation | Less than or equal to 11.0 |
| 9. Steam Generator Water Level--Low-Low | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| b. Turbine Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 10. 4160 volt Emergency Bus Loss of Voltage | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 11. Loss of Main Feedwater Pumps | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 12. Reactor Coolant Pump Bus Undervoltage | |
| a. Turbine Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |

TABLE NOTATION

# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

THIS PAGE INTENTIONALLY LEFT BLANK.

THIS PAGE INTENTIONALLY LEFT BLANK.

DESIGN FEATURES

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal $T_{avg}$ of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A $k_{eff}$ equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks.

c.1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel with nominal enrichment above 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU. In Region 1, fuel shall be stored in a three-out-of-four cell configuration with one symmetric cell location of each 2 x 2 cell array vacant.

2. The boundary between the Region 1 mentioned above and the rest of the spent fuel storage racks (defined as Region 2) shall be such that the three-out-of-four storage requirement shall be carried into Region 2 by, at least, one row as shown in Figure 5.6-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T") is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steam line break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the reactor protection system.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

## 3/4.3 INSTRUMENTATION

## BASES

### 3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes @ and @@) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note ++), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumption used for VCT and RWST valves are valid.

## 3/4.3.3 MONITORING INSTRUMENTATION

### 3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be OPERABLE at all times during plant operation, but they are not required for safe shutdown of the plant.

In addition, a minimum of two in containment radiation-level monitors with a maximum range of $10^7$ R/hr for photon only should be OPERABLE at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible.

Table 3.3-6 is based on the following Alarm/Trip Setpoints and Measurement Ranges for each instrument listed. For the unit vent noble gas monitors, it should be noted that there is an automatic switchover from the low/mid-range channels to the high-range channel when the upper limits of the low- and mid-range channel measurement ranges are reached. In this case there is no flow to the low- and mid-range channels from the unit vent sample line. This is considered to represent proper operation of the monitor. Therefore, if automatic switchover to the high-range should occur, and the low- and mid-range detectors are capable of functioning when flow is re-established, the low- and mid-range channels should not be declared inoperable and the ACTION statement in the Technical Specification does not apply. This is also true while purging the low- and mid-range chambers following a large activity excursion prior to resumption of low-level monitoring and establishment of a new background.

| INSTRUMENT | ALARM/TRIP SETPOINT | MEASUREMENT RANGE* |
|---|---|---|
| 1) Area Monitor-Upper Containment (VRS 1101/1201) | The monitor trip setpoint is based on 10 CFR 20 limits. A homogeneous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate. | $10^{-4}$R/hr to 10R/hr. |

* This is the minimum required sensitivity of the instrument. Indicated values on these instruments above or below these minimum sensitivity ranges are acceptable and indicate existing conditions not instrument inoperability.

Radiation Monitoring Instrumentation (Continued)

| INSTRUMENT | ALARM/TRIP SETPOINT | MEASUREMENT RANGE* |
|---|---|---|
| 2) Area Monitor Containment High Range (VRA 1310/ 1410) | The monitor setpoint was selected to reflect the guidance provided in Generic Letter 83-37 for NUREG-0737 Technical Specifications | 1R/hr to $1 \times 10^7$ R/hr Photons. |
| 3) Process Monitor Particulate (ERS 1301/1401) | The monitor trip setpoint is based on 10 CFR 20 limits. The setpoint was determined using the Noble gas setpoint and historical monitor data of the ratio of particulates to Noble gases. | $1.5 \times 10^{-4}$ uCi to 7.5 uCi |
| 4) Process Monitor Noble Gas (ERS 1305/1405) | The monitor trip setpoint is based on 10 CFR 20 limits. A homogeneous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate. | $5.8 \times 10^{-7}$ uCi/cc to $2.7 \times 10^{-2}$ uCi/cc |
| 5) Steam Generator PORV (MRA 1601) (MRA 1602) (MRA 1701) (MRA 1702) | Not Applicable.** | 0.1uCi/cc to $1.0 \times 10^2$ uCi/cc. |

* This is the minimum sensitivity of the instrument for normal operation, to follow the course of an accident, and/or take protective actions. Values of the instrument above or below this minimum sensitivity range are acceptable.

** These monitors are used to provide data to assist in post-accident off-site dose assessment.

INSTRUMENTATION

BASES

Radiation Monitoring Instrumentation (Continued)

| INSTRUMENT | ALARM/TRIP SETPOINT | MEASUREMENT RANGE* |
| --- | --- | --- |
| 6) Noble Gas Unit Vent Monitors | | |
| a) Low Range (VRS 1505) | See Bases Section 3/4.3.3.10 | $5.8x10^{-7}$ uCi/cc to $2.7x10^{-2}$ uCi/cc |
| b) Mid Range (VRS 1507) | Not Applicable** | $1.3x10^{-3}$ uCi/cc to $7.5x10^{2}$ uCi/cc |
| c) High Range (VRS 1509) | Not Applicable** | $2.9x10^{-2}$ uCi/cc to $1.6x10^{4}$ uCi/cc |
| 7) Gland Steam Condenser Vent Noble Gas Monitor | | |
| a) Low Range (SRA 1805) | See Bases Section 3/4.3.3.10 | $5.8x10^{-7}$ uCi/cc to $2.7x10^{-2}$ uCi/cc |
| 8) Steam Jet Air Ejector Vent Noble Gas Monitor | | |
| a) Low Range (SRA 1905) | See Bases Section 3/4.3.3.10 | $5.8x10^{-7}$ uCi/cc to $2.7x10^{-2}$ uCi/cc. |
| b) Mid Range (SRA 1907) | Not applicable.** | $1.3x10^{-3}$ uCi/cc to $7.5x10^{2}$ uCi/cc. |
| c) High Range (SRA 1909) | Not Applicable.** | $2.9x10^{-2}$ uCi/cc to $1.6x10^{4}$ uCi/cc. |
| 9) Spent Fuel Storage (RRC-330) | The monitor setpoint is selected to alarm and trip consistent with 10 CFR 70.24(a) (2) | $1x10^{-1}$ mR/hr to $1x10^{4}$ mR/hr |

* This is minimum sensitivity of the instrument for normal operation, to follow the course of an accident, and/or take protective actions. Values of the instrument above or below this minimum sensitivity range are acceptable.

** These monitors are used to provide data to assist in post-accident off-site dose assessment.

Radiation Monitoring Instrumentation (Continued)

The Radiation Monitoring Instrumentation Surveillance Requirements per Table 4.3-3 are based on the following interpretation:

1) The CHANNEL FUNCTIONAL TEST is successfully accomplished by the injection of a simulated signal into the channel, as close to the detector as practical, to verify the channel's alarm and/or trip function only.

2) The CHANNEL CALIBRATION as defined in T/S Section 1.9 permits the "known values" generated from radioactive calibration sources to be supplemented with "known values" represented by simulated signals for that subset of "known values" required for calibration and not practical to generate using the radioactive calibration sources.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. For the meteorological instrumentation, the required channel check consists of a qualitative assessment of channel behavior during operation by observation. For the 10 m wind speed and wind direction instruments the channel check also includes, when possible, a comparison of channel indications.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Appendix R remote shutdown instrumentation ensures that sufficient instrumentation is available to permit shutdown of the facility to COLD SHUTDOWN conditions at the local shutdown indication (LSI) panel. In the event of a fire, normal power to the LSI panels may be lost. As a result, capability to repair the LSI panels from Unit 2 has been provided. If the alternate power supply is not available, fire watches will be established in those fire areas where loss of normal power to the LSI panels could occur in the event of fire. This will consist of either establishing continuous fire watches or verifying OPERABILITY of fire detectors per Specification 4.3.3.7 and establishing hourly fire watches. The details of how these fire watches are to be implemented are included in a plant procedure.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION (SYSTEMS/DETECTORS)

OPERABILITY of the fire detection systems/detectors ensures that adequate detection capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of the fires will reduce the potential for damage to safety related systems or components in the areas of the specified systems and is an integral element in the overall facility fire protection program. In the event that a portion of the fire detection systems is inoperable, the ACTION statements provided maintain the facility's fire protection program and allows for continued operation of the facility until the inoperable system(s)/detector(s) are restored to OPERABILITY. However, it is not our intent to rely upon the compensatory action for an extended period of time and action will be taken to restore the minimum number of detectors to OPERABLE status within a reasonable period.

3/4.3.3.8 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

The containment water level and containment sump level transmitters will be modified or replaced and OPERABLE* by the end of the refueling outage to begin in February 1989.

*Amendment 112 (Effective before startup following refueling outage currently scheduled in 2/89).

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

3/4.3.3.9 The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

3/4.3.3.10 The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

## BASES

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 170°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine $F_Q$ limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of $70^{\circ}F$. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. The value of the minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to $80^{\circ}F$ to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure from containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 142
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated March 26, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 142 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective prior to restart from the next refueling outage but no later than August 31, 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

for L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 20, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 142

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| VII | VII |
| XII | XII |
| 3/4 3-9 | 3/4 3-9 |
| 3/4 3-26 | 3/4 3-26 |
| 3/4 3-27 | 3/4 3-27 |
| 3/4 3-28 | 3/4 3-28 |
| 3/4 3-29 | 3/4 3-29 |
| 3/4 5-9 | 3/4 5-9 |
| 3/4 5-10 | 3/4 5-10 |
| B 2-5 | B 2-5 |
| B 3/4 3-1a | B 3/4 3-1a |
| B 3/4 3-1b | B 3/4 3-1b |
| B 3/4 3-1c | B 3/4 3-1c |
| B 3/4 3-1d | B 3/4 3-1d |
| B 3/4 5-2 | B 3/4 5-2 |
| B 3/4 5-3 | B 3/4 5-3 |

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## INDEX

BASES

## TABLE 3.3-2
## REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| 1. Manual Reactor Trip | NOT APPLICABLE |
| 2. Power Range, Neutron Flux (High and Low Setpoint) | Less than or equal to 0.5 seconds* |
| 3. Power Range, Neutron Flux, High Positive Rate | NOT APPLICABLE |
| 4. Power Range, Neutron Flux High Negative Rate | Less than or equal to 0.5 seconds* |
| 5. Intermediate Range, Neutron Flux | NOT APPLICABLE |
| 6. Source Range, Neutron Flux | NOT APPLICABLE |
| 7. Overtemperature Delta T | Less than or equal to 6.0 seconds* |
| 8. Overpower Delta T | Less than or equal to 6.0 seconds* |
| 9. Pressurizer Pressure--Low | Less than or equal to 2.0 seconds |
| 10. Pressurizer Pressure--High | Less than or equal to 2.0 seconds |
| 11. Pressurizer Water Level--High | Less than or equal to 2.0 seconds |

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

## TABLE 3.3-5

## ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | | RESPONSE TIME IN SECONDS |
|---|---|---|
| 1. Manual | | |
| a. | Safety Injection (ECCS) | Not Applicable |
| | Feedwater Isolation | Not Applicable |
| | Reactor Trip (SI) | Not Applicable |
| | Containment Isolation-Phase "A" | Not Applicable |
| | Containment Purge and Exhaust Isolation | Not Applicable |
| | Auxiliary Feedwater Pumps . | Not Applicable |
| | Essential Service Water System | Not Applicable |
| b. | Containment Spray | Not Applicable |
| | Containment Isolation-Phase "B" | Not Applicable |
| | Containment Purge and Exhaust Isolation | Not Applicable |
| | Containment Air Recirculation Fan | Not Applicable |
| c. | Containment Isolation-Phase "A" | Not Applicable |
| | Containment Purge and Exhaust Isolation | Not Applicable |
| d. | Steam Line Isolation | Not Applicable |
| 2. Containment Pressure-High | | |
| a. | Safety Injection (ECCS) | Less than or equal to 27.0@@/27.0++ |
| b. | Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. | Feedwater Isolation | Less than or equal to 8.0 |
| d. | Containment Isolation-Phase "A" | Not Applicable |
| e. | Containment Purge and Exhaust Isolation | Not Applicable |
| f. | Auxiliary Feedwater Pumps | Not Applicable |
| g. | Essential Service Water System | Not Applicable |

## TABLE 3.3-5 (Continued)

## ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| **3. Pressurizer Pressure-Low** | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0@@/27.0++ |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation - Phase "A" | Less than or equal to 18.0# |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| g. Essential Service Water System | Less than or equal to 48.0++/13.0# |
| **4. Differential Pressure Between Steam Lines - High** | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0@@/37.0@ |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation - Phase "A" | Less than or equal to 18.0#/28.0## |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| g. Essential Service Water System | Less than or equal to 13.0#/48.0## |
| **5. Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low** | |
| a. Safety Injection (ECCS) | Not Applicable |
| b. Reactor Trip (from SI) | Not Applicable |
| c. Feedwater Isolation | Not Applicable |
| d. Containment Isolation-Phase "A" | Not Applicable |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Auxiliary Feedwater Pumps | Not Applicable |
| g. Essential Service Water System | Not Applicable |
| h. Steam Line Isolation | Less than or equal to 13.0 |

TABLE 3.3-5 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 6. Steam Line Pressure--Low | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0@@/37.0@ |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation-Phase "A" | Less than or equal to 18.0#/28.0## |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| g. Essential Service Water System | Less than or equal to 14.0#/48.0## |
| h. Steam Line Isolation | Less than or equal to 11.0 |
| 7. Containment Pressure--High-High | |
| a. Containment Spray | Less than or equal to 45.0 |
| b. Containment Isolation-Phase "B" | Not Applicable |
| c. Steam Line Isolation | Less than or equal to 10.0 |
| d. Containment Air Recirculation Fan | Less than or equal to 600.0 |
| 8. Steam Generator Water Level--High-High | |
| a. Turbine Trip | Less than or equal to 2.5 |
| b. Feedwater Isolation | Less than or equal to 11.0 |
| 9. Steam Generator Water Level--Low-Low | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| b. Turbine Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 10. 4160 volt Emergency Bus Loss of Voltage | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 11. Loss of Main Feedwater Pumps | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 12. Reactor Coolant Pump Bus Undervoltage | |
| a. Turbine Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |

TABLE 3.3-5 (Continued)

TABLE NOTATION

# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

THIS PAGE INTENTIONALLY LEFT BLANK.

THIS PAGE INTENTIONALLY LEFT BLANK.

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T'') is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. The overpower delta T reactor trip provides protection or back-up protection for at-power steam line break events. Credit was taken for operation of this trip in the steam line break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the uncontrolled control rod assembly bank withdrawal at-power event.

BASES

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes @ and @@) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note ++), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumption used for VCT and RWST valves are valid.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be OPERABLE at all times during plant operation, but they are not required for safe shutdown of the plant.

In addition, a minimum of two in containment radiation-level monitors with a maximum range of $10^7$ R/hr for photon only should be OPERABLE at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible.

Table 3.3-6 is based on the following Alarm/Trip Setpoints and Measurement Ranges for each instrument listed. For the unit vent noble gas monitors, it should be noted that there is an automatic switchover from the low/mid-range channels to the high-range channel when the upper limits of the low- and mid-range channel measurement ranges are reached. In this case there is no flow to the low- and mid-range channels from the unit vent sample line. This is considered to represent proper operation of this monitor. Therefore, if automatic switchover to the high-range should occur, and the low- and mid-range detectors are capable of functioning when flow is re-established, the low- and mid-range channels should not be declared inoperable and the ACTION statement in the Technical Specification does not apply. This is also true while purging the low- and mid-range chambers following a large activity excursion prior to resumption of low-level monitoring and establishment of a new background.

Radiation Monitoring Instrumentation (Continued)

| INSTRUMENT | ALARM/TRIP SETPOINT | MEASUREMENT RANGE* |
|---|---|---|
| 1) Area Monitor-Upper Containment (VRS 2101/2201) | The monitor trip setpoint is based on 10 CFR 20 limits. A homogenous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate. | $10^{-4}$R/hr to 10R/hr. |
| 2) Area Monitor-Containment High Range (VRA 2310/2410) | The monitor setpoint was selected to reflect the guidance provided in Generic Letter 83-37 for NUREG-0737 Technical Specifications. | 1R/hr to 1 x $10^{7}$R/hr Photons. |
| 3) Process Monitor Particulate (ERS 2301/2401) | The monitor trip setpoint is based on 10 CFR 20 limits. The setpoint was detemined using the Noble gas setpoint and historical monitor data of the ratio of particulate to Noble gases. | 1.5x$10^{-4}$uCi to 7.5 uCi. |
| 4) Process Monitor Noble Gas (ERS 2305/2405) | The monitor trip setpoint is based on 10 CFR 20 limits. A homogenous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate. | 5.8x$10^{-7}$uCi/cc to 2.7x$10^{-2}$uCi/cc |
| 5) Steam Generator PORV (MRA 2601) (MRA 2602) (MRA 2701) (MRA 2702) | Not Applicable.** | 0.1uCi/cc to 1.0x$10^{2}$ uCi/cc. |

*This is the minimum required sensitivity of the instrument. Indicated values on these instruments above or below these minimum sensitivity ranges are acceptable and indicate existing conditions not instrument inoperability.

**These monitors are used to provide data to assist in post-accident off-site dose assessment.

INSTRUMENTATION

BASES

Radiation Monitoring Instrumentation (Continued)

| INSTRUMENT | ALARM/TRIP SETPOINT | MEASUREMENT RANGE* |
|---|---|---|
| 6) Noble Gas Unit Vent Monitors | | |
| a) Low Range (VRS 2505) | See Bases Section 3/4.3.3.10 | $5.8 \times 10^{-7}$ uCi/cc to $2.7 \times 10^{-2}$ uCi/cc. |
| b) Mid Range (VRS 2507) | Not Applicable** | $1.3 \times 10^{-3}$ uCi/cc to $7.5 \times 10^{2}$ uCi/cc |
| c) High Range (VRS 2509) | Not Applicable** | $2.9 \times 10^{-2}$ uCi/cc to $1.6 \times 10^{4}$ uCi/cc |
| 7) Gland Steam Condenser Vent Noble Gas Monitor | | |
| a) Low Range (SRA 2805) | See Bases Section 3/4.3.3.10 | $5.8 \times 10^{-7}$ uCi/cc to $2.7 \times 10^{-2}$ uCi/cc. |
| 8) Steam-Jet Air Ejector Vent Noble Gas Monitor | | |
| a) Low Range (SRA 2905) | See Bases Section 3/4.3.3.10 | $5.8 \times 10^{-7}$ uCi/cc to $2.7 \times 10^{-2}$ uCi/cc. |
| b) Mid Range (SRA 2907) | Not applicable.** | $1.3 \times 10^{-3}$ uCi/cc to $7.5 \times 10^{2}$ uCi/cc. |
| c) High Range (SRA 2909) | Not Applicable.** | $2.9 \times 10^{-2}$ uCi/cc to $1.6 \times 10^{4}$ uCi/cc. |
| 9) Spent Fuel Storage (RRC-330) | The monitor setpoint is selected to alarm and trip consistent with 10 CFR 70.24(a) (2) | $1 \times 10^{-1}$ mR/hr to $1 \times 10^{4}$ mR/hr |

* This is minimum sensitivity of the instrument for normal operation, to follow the course of an accident, and/or take protective actions. Values of the instrument above or below this minimum sensitivity range are acceptable.

** These monitors are used to provide data to assist in post-accident off-site dose assessment.

INSTRUMENTATION

BASES

The Radiation Monitoring Instrumentation Surveillance Requirements per Table 4.3-3 are based on the following interpretation:

1) The CHANNEL FUNCTIONAL TEST is successfully accomplished by the injection of a simulated signal into the channel, as close to the detector as practical, to verify the channel's alarm and/or trip function only.

2) The CHANNEL CALIBRATION as defined in T/S Section 1.9 permits the "known values" generated from radioactive calibration sources to be supplemented with "known values" represented by simulated signals for that subset of "known values" required for calibration and not practical to generate using the radioactive calibration sources.

## EMERGENCY CORE COOLING SYSTEMS

## BASES

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or steam line rupture. The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine $F_Q$ limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of $80^{\circ}F$. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.