LICENSE AMENDMENT REQUEST DATED December 14, 1995

Conformance of Administrative Controls Section 6
to the Guidance of Standard Technical Specifications

EXHIBIT B

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# APPENDIX A TECHNICAL SPECIFICATIONS

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3.1.D. MAXIMUM COOLANT ACTIVITY

1. The specific activity of the primary coolant (except as specified in 3.1.D.2 and 3 below) shall be limited to:

   a. Less than or equal to 1.0 microcuries per gram DOSE EQUIVALENT I-131, and

   b. Less than or equal to $100/\bar{E}$ microcuries per gram of gross radioactivity.

2. If a reactor is critical or the reactor coolant system average temperature is greater than or equal to 500°F:

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure TS.3.1-3, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

   b. With the specific activity of the primary coolant greater than 100/E microcurie per gram, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

3. If a reactor is at or above COLD SHUTDOWN, with the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/E microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.1-2B until the specific activity of the primary coolant is restored to within its limits.

~~4. Annual reporting requirements are identified in 6.7.A.1.e.~~

## 3.1.E. MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

1. Concentrations of contaminants in the reactor coolant as measured per Specification 4.1.C shall not exceed the following limits when the reactor coolant is above 250°F

| Contaminant | Steady State Limits (PPM) | Transient Limits (PPM) |
|---|---|---|
| a. Oxygen | 0.10 | 1.00 |
| b. Chloride | 0.15 | 1.50 |
| c. Fluoride | 0.15 | 1.50 |

2. If any of the steady state limits as specified in 3.1.E.1 above are determined to be exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken.

3. If the concentrations of any of the contaminants cannot be reduced below the steady state limits of Specification 3.1.E.1 above in a 24 hour period, or the transient limits are reached, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours, and take corrective actions. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the transient limits. Otherwise, a safety review per Specification 6.2 is required before startup.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is below 250°F.

| Contaminant | Steady State Limits (PPM) | Transient Limits (PPM) |
|---|---|---|
| a. Oxygen | Saturated | Saturated |
| b. Chloride | 0.15 | 1.5 |
| c. Fluoride | 0.15 | 1.5 |

If the steady state limits above are exceeded, and the concentrations cannot be reduced below the steady state limits in a 48 hour period, the reactor shall be brought to COLD SHUTDOWN and corrective action taken.

5. For the purposes of correcting the contaminant concentrations to meet Specifications 3.1.E.1 and 3.1.E.4 above, operation of reactor coolant pumps for a short period of time to assure mixing of the coolant shall be permitted during COLD SHUTDOWN, provided the coolant temperature does not exceed 250°F.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

| TEST | FREQUENCY |
|---|---|
| 1. RCS Gross Activity Determination | 5/week |
| 2. RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | 1/14 days (when at power) |
| 3. RCS Radiochemistry $\bar{E}$ determination | 1/6 months(1) (when at power) |
| 4. RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135 | a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and<br>b) One sample between 2 and 6 hours following thermal POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period ( above hot shutdown) |
| 5. RCS Radiochemistry (2) | Monthly |
| 6. RCS Tritium Activity | Weekly |
| 7. Deleted~~RCS Chemistry (Cl*, F*, O2)~~ | ~~5/Week~~ |
| 8. RCS Boron Concentration*(3) | 2/Week (4) |
| 9. RWST Boron Concentration | Weekly |
| 10. Boric Acid Tanks Boron Concentration | 2/Week |
| 11. Caustic Standpipe NaOH Concentration | Monthly |
| 12. Accumulator Boron Concentration | Monthly |
| 13. Spent Fuel Pit Boron Concentration | Monthly/Weekly(7)(8) |

~~* Required at all times.~~

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.

C. Containment Vacuum Breakers

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containmant vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested during each refueling shutdown in accordance with the requirements of Specification 4.4.A.2.

~~D. Residual Heat Removal System~~

~~1. Those portions of the residual heat removal system external to the isolation valves at the containment, shall be hydrostatically tested for leakage during each refueling shutdown.~~

~~2. Visual inspection shall be made for excessive leakage from components of the system. Any visual leakage that cannot be stopped at test conditions shall be measured by collection and weighing or by another equivalent method.~~

~~3. The acceptance criterion is that maximum allowable leakage from either train of the recirculation heat removal system components (which includes valve stems; flanges and pump seals) shall not exceed two gallons per hour when the system is at 350 psig.~~

~~4. Repairs shall be made as required to maintain leakage within the acceptance criterion in Specification 4.4.D.3~~

~~5. If repairs are not completed within 7 days, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion in 3. above is satisfied.~~

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power sources and equipment are OPERABLE.

Specification

The following tests and surveillance shall be performed:

A. Diesel Generators

1. At least once each month, for each diesel generator:

   a. Verify the fuel level in the day tank.

   b. Verify the fuel level in the fuel storage tank.

   c. Deleted~~Verify that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment.~~

   d. Verify the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

   e. Verify the diesel generator can start and gradually accelerate. Verify the generator voltage and frequency can be adjusted to 4160 ± 420 volts and 60 ± 1.2 Hz. Subsequently, manually sychronize the generator, gradually load to at least 1650 kW (Unit 2: 5100 kW to 5300 KW), and operate for at least 60 minutes. This test should be conducted in consideration of the manufacturer's recommendations regarding engine prelube, warm-up, loading and shutdown procedures where possible.

# 5.0 DESIGN FEATURES

## 5.1 SITE

The Prairie Island Nuclear Generating Plant is located on property owned by Northern States Power (NSP) Company at a site on the west bank of the Mississippi River, approximately 6 miles northwest of the city of Red Wing, Minnesota. The minimum distance from the center line of either reactor to the site exclusion boundary is 715 meters, and the low population zone distance is 1-1/2 miles. The nearest population center of 25,000 or more people is South Saint Paul. These site characteristics comply with definitions in 10CFR100 (Reference 1).

The U.S. Army Corp of Engineers controls the land within the exclusion area that is not owned by NSP. The Corps has made an agreement with NSP to prevent residential construction on this land for the life of the plant (Reference 2).

These specifications use atmospheric diffusion factors based on the NRC staff evaluations. Its evaluation of accidental airborne releases is based on a relative concentration of $9.8 \times 10^{-4}$ seconds per cubic meter at the site boundary. Its evaluation of routine releases is based on a relative concentration of $1.5 \times 10^{-5}$ seconds per cubic meter (Reference 3).

The flood of record in 1965 produced a water surface elevation of +688 feet MSL at the site. The calculated probable maximum flood (PMF) level is +703.6 feet mean sea level (MSL), and the estimated wave runup could reach +706.7 feet MSL. (See Section 2.4.2 of this report.) Plant grade level is +695 feet MSL.

Flood protection structures have been provided. The two turbine support facilities, the common auxiliary building, and the two shield buildings have been physically connected by a concrete flood wall, most of the length of which constitutes the concrete foundation walls for the various buildings. The top of this wall supports the metal siding for the buildings at about elevation +705 feet MSL. Fourteen doors through the flood wall, or into the various buildings (including the separate screen house), are provided with receivers for the erection of flood protection panels to prevent flood water from reaching safety related facilities.

The cooling water pumps in the screenhouse are designed to operate up to a flood level of +695 feet MSL without flood protection measures, and up to a level of +707 feet MSL with the erection of flood protection panels. The main transformer foundation is at +695 feet MSL. The transformer will function to a flood level of+698 feet MSL.

~~The Technical Specification 6.5 A.7 requires an emergency procedure that will necessitate plant shutdown for flood water levels above +692 feet MSL at the plant site. The emergency procedure will assure the~~

~~proper erection of flood protection panels and assure an orderly shutdown of the plant and protection of safety related facilities. This procedure will provide for progressive action levels to prevent the possibility of unsafe plant operation and will include requirements for periodic inspection of flood protection measures.~~

The plant is designed for a design basis earthquake having a horizontal ground acceleration of 0.12g and an operational basis earthquake having a horizontal ground acceleration of 0.06g. ~~An emergency procedure will be prepared in accordance with Specification 6.5.A.7 to define actions required for earthquakes, including plant shutdown and inspection if an operational basis earthquake is measured at the site.~~

References

1. USAR, Section 2.2.1
2. USAR, Section 3.4.5
3. SER, Sections 2.3.4 and 2.3.5

6.0 ADMINISTRATIVE CONTROLS

6.1 ~~Organization~~Responsibility

A. The ~~P~~plant ~~M~~manager shall be responsible for overall unit ~~safe~~ operation and shall ~~have control over those onsite activities necessary for safe operation and maintenance of the plant. The Plant Manager shall~~ delegate in writing the succession to this responsibility during his absence. ~~The Plant Manager has the responsibility for the Fire Protection Program.~~

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

B. The shift supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active senior reactor operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator license shall be designated to assume the control room command function.

## 6.2 Organization

### ~~A~~B. Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions ~~responsible~~ for activities affecting ~~plant~~ safety of the nuclear power plant.

1.-Lines of authority, responsibility and communication shall be defined and established throughout ~~and defined for the~~ highest management levels, ~~through~~ intermediate levels, ~~to~~ and ~~including~~ all operating organization positions. These relationships shall be documented and updated, as appropriate, in ~~the form of~~ organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in ~~the Operational Quality Assurance Plan or~~ the Updated Safety Analysis Report.

2.~~3. There shall be an individual management position~~ The ~~(P~~plant ~~M~~manager~~) in the onsite organization having responsibility~~ shall report to the corporate vice president specified in 6.2.A.3, shall be responsible for overall ~~unit~~ safe operation of the plant, and ~~who~~ shall have control over those onsite activities ~~resources~~ necessary for safe operation and maintenance of the plant.

3~~2~~. A corporate ~~There shall be an individual executive position (V~~vice ~~P~~president ~~Nuclear Generation) in the offsite organization having~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff, ~~and those who~~ carry out health physics, or perform ~~and~~ quality assurance functions may report to the appropriate onsite manager; however, these individuals ~~they~~ shall have sufficient organizational freedom to ensure their independence from operating pressures.

### B~~C~~. Plant Staff~~:~~

The plant staff organization shall include the following:

1. An operator to perform non-licensed duties shall be assigned to each reactor containing fuel and one additional operator to perform non-licensed duties shall be assigned when either or both reactors are operating in MODES 1, 2, 3, or 4. Also, if one unit is in MODE 1, 2, 3, or 4 and the other unit is in MODE 5 or 6, as a minimum the on-site staffing shall include two senior reactor operators(SRO) and two licensed reactor operators(RO).

~~Each on duty shift shall be composed of at least the minimum shift crew composition shown on Table TS.6.1-1.~~

2. At least one licensed operator shall be present in the control room for each reactor containing fuel. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed senior reactor operator shall be present in the control room.

   ~~For each reactor that contains fuel: a licensed operator in the control room.~~

3. Shift crew composition may be less than the minimum requirement of 10CFR50.54(m)(2)(i) and 6.2.B.1 and 6.2.B.7 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

   ~~At least two licensed operators shall be he present in the control room during a reactor startup, a scheduled reactor shutdown, and during recovery from a reactor trip. These operators are in addition to those required for the other reactor.~~

4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. The amount of overtime worked by plant staff members performing safety related functions shall be limited and controlled by procedures which implement an NRC approved program.

   ~~All refueling operations shall be directly supervised by a licensed Senior Reactor Operator or a Senior Reactor Operator Limited to Fuel Handling who has no other concurrent respons-ibilities during this operation.~~

6. The ~~General Superintendent Plant~~ Ooperations manager or assistant operations manager shall ~~be formerly licensed or~~ hold ~~a current~~ an SRO license ~~on a similar type plant~~.

~~7. At least one member of plant management holding a current Senior Reactor Operator license shall be assigned to the plant operations group on a long term basis (approximately two years). This individual shall not be assigned to a rotating shift.~~

7. The shift technical advisor (STA) shall provide advisory technical support to the shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. Personnel performing the function of the STA shall be assigned to the shift crew when a unit is in MODE 1, 2, 3, or 4.

~~D.~~ [Paragraph 6.1.D was relocated to provide the substance for new Section 6.3

~~E. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:~~

~~1. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:~~

~~a. An individual should not be permitted to work more than 16 hours straight excluding shift turnover time.~~

~~b. Overtime should be limited for all nuclear plant staff personnel so so that total work time does not exceed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.~~

~~c. A break of at least eight hours including shift turnover time should be allowed between work periods.~~

~~d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~

~~e. Shift Emergency Coordinator (SEC) on-site rest time periods shall not be considered as hours worked when determining the total work time for which the above limitations apply~~

~~2. Any deviation from the above guidelines shall be authorized by the Plant Manager or designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. During plant emergencies, the Emergency Director shall have this authority. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not allowed.~~

~~TABLE TS.6.1-1~~

~~MINIMUM SHIFT CREW COMPOSITION (Note 1 and 3)~~

| ~~CATEGORY~~ | ~~BOTH UNITS IN COLD SHUTDOWN OR REFUELING SHUTDOWN~~ | ~~ONE UNIT IN COLD SHUTDOWN OR REFUELING SHUTDOWN AND ONE UNIT ABOVE COLD SHUTDOWN~~ | ~~BOTH UNITS ABOVE COLD SHUTDOWN~~ |
|---|---|---|---|
| ~~No. Licensed Senior Operators (LSO)~~ | ~~2 (Note 2)~~ | ~~2 (Notes 2, 4)~~ | ~~2 (Note 4)~~ |
| ~~Total No. Licensed Operators (LSO & LO)~~ | ~~4~~ | ~~4~~ | ~~5~~ |
| ~~Total No. Licensed & Unlicensed Operators~~ | ~~6~~ | ~~7~~ | ~~8~~ |
| ~~Shift Manager (Note 5)~~ | ~~0~~ | ~~1~~ | ~~1~~ |

~~NOTES:~~

1. ~~Shift crew composition may be one less than the minimum requirements for a period of time not to exceed two hours in order to accommodate an unexpected absence of one duty shift crew member provided immediate action is taken to restore the shift crew composition to within the minimum requirements specified.~~

2. ~~Does not include the licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations.~~

3. ~~Each LSO and LO shall be licensed on each unit.~~

4. ~~One LSO shall be in the control room at all times when a reactor is above cold shutdown.~~

5. ~~The Shift Manager performs the functions of the Shift Technical Advisor~~

~~6.2 Review and Audit~~

~~Organizational units for the review and audit of facility operations shall be constituted and have the responsibilities and authorities outlined below:~~

~~A. Safety Audit Committee (SAC)~~

~~The Safety Audit Committee provides the independent review of plant operations from a nuclear safety standpoint. Audits of plant operation are conducted under the cognizance of the SAC.~~

~~1. Membership~~

~~a. The SAC shall consist of at least five (5) persons.~~

~~b. The SAC chairman shall be an NSP representative, not having line responsibility for plant operation, appointed by the Vice President Nuclear Generation. Other SAC members shall be appointed by the Vice President Nuclear Generation or by such other person as he may designate. The Chairman shall appoint a Vice Chairman from the SAC membership to act in his absence.~~

~~c. No more than two members of the SAC shall be from groups holding line responsibility for operation of the plant.~~

~~d. A SAC member may appoint an alternate to serve in his absence, with concurrence of the Chairman. No more than one alternate shall serve on the SAC at any one time. The alternate member shall have voting rights.~~

~~2. Qualifications~~

~~a. The SAC members should collectively have the capability required to review activities in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, quality assurance practices, and other appropriate fields associated with the unique characteristics of the nuclear power plant.~~

~~b. When the nature of a particular problem dictates, special consultants will be utilized, as necessary, to provide expert advice to the SAC.~~

~~3. Meeting Frequency~~

~~The SAC shall meet on call by the Chairman but not less frequently than twice a year.~~

~~4. Quorum~~

~~a. No less than a majority of the permanent members or their alternates, including the SAC Chairman or Vice Chairman.~~

~~b. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.~~

~~5. Responsibilities - The following subjects should be reported to or reviewed by the SAC:~~

~~a. Written safety evaluations of (1) changes in the facility, (2) changes to procedures, and (3) tests or experiments completed without prior approval under the provisions of 10 CFR 50.59 to verify that such changes, tests or experiments did not involve a change in the Appendix A Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59.~~

~~b. Proposed changes to procedures, changes in the facility, tests and experiments which may involve a change in the Appendix A technical specifications or an unreviewed safety question as defined in 10 CFR 50.59. Matters of this kind shall be referred to the SAC following their review by the onsite operating organization.~~

~~c. Proposed changes in Appendix A Technical Specifications or proposed license amendments relating to nuclear safety.~~

~~d. Violations of applicable codes, regulations, orders, Appendix A Technical Specifications, and license requirements or internal procedures or instructions having nuclear safety significance.~~

~~e. Significant operating abnormalities or deviations from normal and expected performance of plant safety-related structures, systems, or components.~~

~~f. Investigation of all Reportable Events and events requiring Special Reports to the Commission.~~

~~g. Revisions to the Facility Emergency Plan, Facility Security Plan, and the Fire Protection Program.~~

~~h. Operations Committee minutes to determine if matters considered by that Committee involve unreviewed or unresolved safety questions.~~

~~i. Other nuclear safety matters referred to the SAC by the Operations Committee, plant management or company management.~~

~~j. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures systems, or components.~~

~~k. Reports of special inspections and audits conducted in accordance with Specification 6.3.~~

~~l. Changes to the Offsite Dose Calculation Manual (ODCM)~~

~~m. Review of investigative reports of unplanned releases of radioactive material to the environs.~~

~~6. Audit The operation of the nuclear power plant shall be audited formally under the cognizance of the SAC to assure safe facility operation.~~

~~a. Audits of selected aspects of plant operation, as delineated in Paragraph 4.4 of ANSI N18.7-1972, shall be performed with a frequency commensurate with their nuclear safety significant and in a manner to assure that an audit of all nuclear safety related activities is completed within a period of two years. The audits shall be performed in accordance with appropriate written instructions and procedures.~~

~~b. Audits of aspects of plant radioactive effluent treatment and radiological environmental monitoring shall be performed as follows:~~

~~1. Implementation of the Offsite Dose Calculation Manual at least once every two years.~~

~~2. Implementation of the Process Control Program for solidification of radioactive wastes at least once every two years.~~

~~3. The Radiological Environmental Monitoring Program and the results thereof, including quality controls, at least once every year.~~

~~c. Periodic review of the audit program should be performed by the SAC at least twice a year to assure its adequacy.~~

~~d. Written reports of the audits shall be reviewed by the Vice President Nuclear Generation, by the SAC at a scheduled meeting, and by members of management having responsibility in the areas audited.~~

~~7. Authority~~

~~The SAC shall be advisory to the Vice President Nuclear Generation.~~

~~8. Records~~

~~Minutes shall be prepared and retained for all scheduled meetings of the Safety Audit Committee. The minutes shall be distributed within one month of the meeting to the Vice President Nuclear Generation, each member of the SAC and others designated by the Chairman. There shall be a formal approval of the minutes.~~

~~9. Procedures~~

~~A written charter for the SAC shall be prepared that contains:~~

~~a. Subjects within the purview of the group.~~

~~b. Responsibility and authority of the group.~~

~~c. Mechanisms for convening meetings.~~

~~d. Provisions for use of specialists or subgroups.~~

~~e. Authority to obtain access to the nuclear power plant operating record files and operating personnel when assigned audit functions.~~

~~f. Requirements for distribution of reports and minutes prepared by the group to others in the NSP organization.~~

~~B. Operations Committee (OC)~~

~~1. Membership~~

~~The Operations Committee shall consist of at least six (6) regular members drawn from the key supervisors of the onsite staff. The key supervisors include, at least, the following positions: Plant Manager, General Superintendent Plant Operations, General Superintendent Plant Maintenance, General Superintendent Radiation Protection and plant engineering supervisors.~~

~~Alternates to the regular members shall be designated in writing by the Chairman to serve on a temporary basis. No more than two alternates shall participate as voting members of the Operations Committee at any one time.~~

~~The Plant Manager shall serve as Chairman of the OC and shall appoint a regular member to act as Chairman in his absence.~~

~~2. Meeting Frequency~~

~~The Operations Committee will meet on call by the Chairman or as requested by individual members and at least monthly.~~

~~3. Quorum~~

~~A majority of the membership, including the Chairman.~~

~~4 Responsibilities The following subjects shall be reviewed by the Operations Committee:~~

~~a. Proposed tests and experiments and their results.~~

~~b. Modifications to plant systems or equipment as described in the Updated Safety Analysis Report and having nuclear safety significance or which involve an unreviewed safety question as defined in Paragraph 50.59 (c), Part 50, Title 10, Code of Federal Regulations.~~

~~c. Proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that will affect nuclear safety as determined by the Plant Manager.~~

~~d. Proposed changes to the Technical Specifications or operating licenses.~~

~~e. All reported or suspected violations of Technical Specifications, operating license requirements, administrative procedures, operating procedures, Results of investigations, including evaluation and recommendation to prevent recurrence will be reported in writing to the Vice President Nuclear Generation and to the Chairman of the Safety Audit Committee.~~

~~f. Investigations of all Reportable Events and events requiring Special Reports to the Commission.~~

~~g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with offsite support groups.~~

~~h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan, and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed initially and periodically with a frequency commensurate with their safety significance but at an interval of not more than two years. Maintenance work requests and their associated procedures shall be reviewed per the requirements of Section 6.2.C.~~

~~i. Special reviews and investigations, as requested by the Safety Audit Committee.~~

~~j. Review of investigative reports of unplanned releases of radioactive material to the environs.~~

~~k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).~~

~~l. The review of safety evaluations, when safety evaluations are required by 10 CFR Part 50, Section 50.59, for procedures or procedure changes to verify that such actions do not constitute an unreviewed safety question.~~

~~m. Fire Protection Program and implementing procedures and the submittal of recommended changes to the Safety Audit Committee.~~

~~5. Authority~~

~~The OC shall be advisory to the Plant Manager. In the event of a disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the Vice President Nuclear Generation and the Chairman of the SAC for review.~~

~~6. Records~~

~~Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the Vice President Nuclear Generation and others designated by the OC Chairman.~~

~~7. Procedures~~

~~A written charter for the OC shall be prepared that contains:~~

~~a. Responsibility and authority of the group~~

~~b. Content and method of submission of presentations to the Operations Committee~~

~~c. Mechanism for scheduling meetings~~

~~d. Provision for meeting agenda~~

~~e. Use of subcommittees~~

~~f. Review and approval, by members, of OC actions~~

~~g. Provision for distribution of minutes~~

~~C. Maintenance Procedures~~

~~1. Maintenance work requests and their associated procedures required by specification 6.5.C and changes thereto, shall be prepared, reviewed and approved. Each procedure or procedure change shall be reviewed by a qualified individual(s) other than the individual(s) who prepared the procedure or procedure change. The appropriate superintendent (as previously designated by the Plant Manager) shall approve these procedures.~~

~~2. Individuals responsible for reviews performed in accordance with the requirements of specification 6.2.C.1 above shall meet or exceed the qualifications described in section 4.2 or 4.4 of ANSI N18.1-1971 and shall be previously designated by the Plant Manager. Such reviews shall include a determination of whether or not additional cross disciplinary review is necessary.~~

~~3. All reviews shall include a determination of the need for a safety evaluation to establish whether or not an unreviewed safety question is involved.~~

6.3 Plant Staff Qualifications ~~SPECIAL INSPECTIONS AND AUDITS~~

~~A. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site Northern States Power Company personnel or an outside fire protection consultant.~~

~~B. An inspection and audit by an outside qualified fire protection consultant shall be performed at intervals no greater than three years.~~

[Current Prairie Island Technical Specifications Paragraph 6.1.D was relocated to provide the substance of new Section 6.3 as follows:]

Each member of the plant staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975 ~~ANSI N18.1-1971 for comparable positions~~, except for (1) ~~the General Superintendent Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Manager~~ personnel who perform the function of shift technical advisor ~~who~~ shall have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and ~~(3)~~ (2) the ~~General Superintendent Plant O~~operations manager who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.2.B.6. ~~6.1.C.7. The training program shall be under the direction of a designated member of Northern States Power management.~~

~~6.4 SAFETY LIMIT VIOLATION~~

[Deletion of this section was proposed in License Amendment Request, "Pressurizer Safety Valves and Main Steam Safety Valves Lift Setting Tolerance Change and Safety Limit Curve Changes", dated May 4, 1995.]

6.~~45 PLANT OPERATING PROCEDURES~~ Procedures

~~Detailed w~~Written procedures ~~including the applicable checkoff lists and instructions, covering areas listed below~~ shall be established, implemented, and maintained covering the following activities: ~~prepared and followed. These procedures and changes thereto, except as specified in TS 6.5.F and G, shall be reviewed by the operations Committee and approved by a member of plant management designated by the Plant Manager.~~

A. ~~Plant Operations~~The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

~~1. Integrated and system procedures for normal startup, operation and shutdown of the reactor and all systems and components involving nuclear safety of the facility.~~

~~2. Fuel handling operations~~

~~3. Actions to be taken to correct specific and foreseen potential or actual malfunction of systems or components including responses to alarms, primary system leaks and abnormal reactivity changes and including follow-up actions required after plant protective system actions have initiated.~~

~~4. Surveillance and testing requirements that could have an effect on nuclear safety.~~

~~5. Implementing procedures of the Facility Emergency Plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity.~~

~~6. Implementing procedures of emergency plans for coping with earthquakes and floods. The flood emergency plan shall require plant shutdown for water levels at the site higher than 692 feet above MSL.~~

B. The emergency operating procedures required to implement the requirements of NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

C. Quality control for effluent and environmental monitoring;

D.~~7. Implementing procedures of the f~~Fire protection program implementation~~.~~; and

E.~~8~~. All programs specified in Specification 6.5.
~~Implementing procedures for the Process Control Program and Offsite Dose Calculation Manual including quality control measures.~~

~~Drills on the procedures specified in A.3. above, shall be conducted as a part of the retraining program.~~

~~B. Radiological~~

~~Radiation control procedures shall be maintained and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10CFR20. This radiation protection program shall be organized to meet the requirements of 10CFR20.~~

[Revision of paragraph 6.5.B.1 was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", dated July 17, 1995. This submittal proposes to move these requirements to a new Specification Section 6.7 entitled, "High Radiation Area"]

~~1. a. Paragraph 20.203 "Caution signs, labels, signals and controls" In lieu of the "Control device" or alarm signal required by paragraph 20.203(c) (2), each high radiation area in which the intensity of radiation is 1000 mRem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (or continuous escort by a qualified person for the purpose of making a radiation survey) and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.~~

~~b. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr, except that doors shall be locked or attended to prevent unauthorized entry into these areas and the keys or devices for locked doors shall be maintained under the administrative control of the Plant Manager.~~

[6.5.B.2 is proposed to be relocated to the new Specification Section 6.5.B below, entitled, "Primary Coolant Sources Outside Containment", with changes as marked up.]

~~3. A program shall be implemented which will ensure the capability to accurately determine the airborne iodine concentration in essential plant areas under accident conditions. This program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for monitoring, and~~

~~c. Provisions for maintenance of sampling and analysis equipment~~

~~A program acceptable to the Commission was described in letters from L.O. Mayer, NSP, to Director of Nuclear Reactor Regulation, dated December 31, 1979 "Lessons Learned Implementation" and March 13, 1980, "1/1/80 Lessons Learned Implementation Additional Information".~~

[6.5.B.4 is proposed to be relocated to new Specification Section 6.5.C below entitled, "Post Accident Sampling", with changes as marked up.]

~~C. Maintenance and Test~~

~~The following maintenance and test procedures will be developed to satisfy routine inspection, preventive maintenance programs, and operating license requirements.~~

~~1. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.~~

~~2. Routine testing of standby and redundant equipment.~~

~~3. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.~~

~~4. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.~~

~~5. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.~~

~~D. Process Control Program (PCP)~~

[Deletion of paragraph 6.5.D, "Process Control Program (PCP)", was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", dated July 17, 1995.]

## 6.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

### A.~~E~~ ~~Offsite Dose Calculation Manual (ODCM)~~Offsite Dose Calculation Manual (ODCM)

[Revision of paragraph 6.5.E, "Offsite Dose Calculation Manual (ODCM)", was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", dated July 17, 1995. This submittal proposes to relocate those proposed requirements to a new Specification Section 6.5.A entitled, "Offsite Dose Calculation Manual (ODCM)"]

### B. Primary Coolant Sources Outside Containment

~~2.~~This ~~A~~ program provides controls ~~shall be implemented~~ to minimize ~~reduce~~ leakage from those portions of systems outside containment that ~~would or~~ could contain highly radioactive fluids during a serious transient or accident to levels as low as practical ~~levels~~. The systems include portions of Residual Heat Removal, Safety Injection and Containment Spray Systems. The ~~This~~ program shall include the following:

1.~~a. Provisions establishing p~~Preventive maintenance and periodic visual inspection requirements, and

2.~~b.~~Integrated leak test requirements for each system at ~~a frequency not to exceed~~ refueling cycle intervals or less.

~~A program acceptable to the Commission was described in letters from L.O Mayer, NSP, to Director of Nuclear Reactor Regulation, dated December 31, 1979 "Lessons Learned implementation" and March 13, 1980, "1/1/80 Lessons Learned Implementation Additional Information".~~

### C~~4~~. Post Accident Sampling

This~~A~~ program provides controls that ~~shall be implemented which will~~ ensure the capability to obtain and analyze reactor coolant, radioactive gases, ~~iodines~~ and particulates in plant gaseous effluents~~,~~ and containment atmosphere samples under accident conditions. The program shall include the following:

1~~a~~.Training of personnel~~,~~;

2~~b~~.Procedures for sampling and analysis~~,~~; and

3~~c~~.Provisions for maintenance of sampling and analysis equipment.

### D. Radioactive Effluent Controls Program

[A new paragraph 6.5.H, "Radioactive Effluent Controls Program", was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", dated July 17, 1995. This submittal proposes to relocate those proposed requirements to a new Specification Section 6.5.D entitled, "Radioactive Effluent Controls Program"]

### E. Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4 cyclic and transient occurrences to ensure that components are maintained within the design limits.

F. (Reserved)

G. (Reserved)

H. (Reserved)

I. (Reserved)

J. Explosive Gas and Storage Tank Radioactivity Monitoring Program

[A new paragraph 6.5.I,"Explosive Gas and Storage Tank Radioactivity Monitoring Program", was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", dated July 17, 1995. This submittal proposes to relocate those proposed requirements to a new Specification Section 6.5.J entitled, "Explosive Gas and Storage Tank Radioactivity Monitoring Program"]

K. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment.

L. Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   a. a change in the Technical Specifications incorporated in the license; or

   b. a change to the USAR or Bases that involves an unreviewed safety question as defined in 10CFR50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

4. Proposed changes that meet the criteria of Specification 6.5.L.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with USAR updates.

~~F. Security~~

~~Procedures shall be developed to implement the requirements of the Security Plan and the Security Contingency Plan. These implementing procedures, with the exception of those non-safety related procedures which govern work activities exclusively applicable to or performed by security personnel, shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager. Security procedures not reviewed by the Operations Committee shall be reviewed and approved by the Superintendent Security.~~

~~G. Temporary Changes to Procedures~~

~~Temporary changes to Operations Committee reviewed procedures described in A,B,C,D,E and F above, which do not change the intent of the original procedure may be made with the concurrence of two members of the unit management staff, at least one of whom holds a Senior Reactor Operator License. Such changes shall be documented, reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager within one month.~~

~~Temporary changes to security procedures not reviewed by the Operations Committee shall be reviewed by two (2) individuals knowledgeable in the area affected by the procedure.~~

## ~~6.6 PLANT OPERATING RECORDS~~

~~A. Records Retained for Five Years~~

~~Records and logs relative to the following items shall be retained for at least five years:~~

1. ~~Normal plant operation including such items as power level, period of operation at each level, fuel exposure and shutdowns.~~

2. ~~Written shift supervisory and reactor logs.~~

3. ~~Periodic checks, inspections, tests and calibrations of components and systems, as related to these Technical Specifications.~~

4. ~~Reviews of changes made to procedures or equipment or reviews of tests and experiments.~~

5. ~~Principal maintenance activities, including inspection, repairs and substitution or replacement of principal items of equipment pertaining to nuclear safety.~~

6. ~~Records of changes to plant procedures and records of special tests and experiments.~~

7. ~~Records of wind speed and direction.~~

8. ~~Records of individual plant staff members showing qualifications, training and re training.~~

9. ~~Reportable Occurrences.~~

~~B. Records Retained for the Life of the Plan~~

~~Records and logs relative to the following items shall be retained for the life of the plant.~~

1. ~~Liquid and airborne radioactive releases to the environs.~~

2. ~~Radiation exposures for all plant personnel, visitor and contract personnel.~~

3. ~~Off-site environmental monitoring surveys.~~

4. ~~Fuel accountability including new and spent fuel inventories and transfers and fuel assembly histories.~~

5. ~~Radioactive shipments.~~

~~6. Plant radiation and contamination surveys.~~

~~7. Changes made to the plant as it is described in the Final Safety Analysis Report, reflected in updated, corrected and as-built drawings.~~

~~8. Cycling beyond normal limits for those components that have been designed to operate safely for a limited number of cycles beyond such limits.~~

~~9. Reactor coolant system in-service inspections.~~

~~10. Minutes of meetings of the Safety Audit Committee.~~

~~11. Records of the service lives of all safety-related snubbers, including the date at which the service life commences and associated installation and maintenance records.~~

6.6~~7~~ Reporting Requirements

The following reports shall be submitted in accordance with 10CFR50.4

~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.~~

A. ~~Routine Reports~~

~~1. Annual Report~~

~~The Annual Report shall be submitted prior to March 1 of each year and shall contain the following reports.~~

~~a.~~ Occupational Exposure Report[(1)]

A tabulation ~~This report shall cover the previous calendar year. The report should tabulate~~ on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10CFR20.2206. The dose assignments to various duty functions may be estimated~~s~~ based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should ~~shall~~ be assigned to specific major work functions. This report shall be submitted by April 30 of each year.

~~b. Report of Safety and Relief Valve Failures and Challenges~~

~~This report shall contain pressurizer safety and relief valve failures and challenges during the past year.~~

~~c. Primary Coolant Iodine Spike Report~~

~~This report shall document the results of specific activity analysis in which the limits specified in 3.1.D.1 were exceeded during the past year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one~~

~~(1) This report supplements the requirements of 10 CFR 20, Section 20.407. If 10 CFR 20, Section 20.407 is revised to include such information, this Specification is unnecessary.~~

~~6.7.A.1.c cont.~~

~~analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

~~2. Startup Report~~

~~A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption of commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

B. Annual Radiological Environmental Monitoring Report

[Revision of paragraphs 6.7.C.1 and 2, "Annual Radiological Environmental Monitoring Report", and, "Environmental Special Reports," was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", Dated July 17, 1995. This submittal proposes to relocate those proposed requirements to the revised Specification Section 6.6.B entitled, "Annual Radiological Environmental Monitoring Report"]

~~6.7.C.1. Annual Radiation Environmental Monitoring Report~~

~~(a) Annual Radiation Environmental Monitoring Reports covering the operation of the program during the previous calendar year shall be submitted prior to May 1 of each year.~~

~~(b) The Annual Radiation Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a~~

~~comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 4.10.B.1. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.~~

~~(c) The Annual Radiation Environmental Monitoring Reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.~~

~~(d) The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensees participation in the Interlaboratory Comparison Program, required by Specification 4.10.C.1.~~

~~2. Environmental Special Reports~~

~~(a) When radioactivity levels in samples exceed limits specified in Table 4.10-3, an Environmental Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 day period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Environmental Special Report as soon as practicable.~~

~~6.7.A.4.~~C.Radioactive Effluent Report

[Revision of paragraph 6.7.A.4,"Annual Radioactive Effluent Report", was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", Dated July 17, 1995. This submittal proposes to relocate those proposed requirements to the revised Specification Section 6.6.C entitled, "Radioactive Effluent Report"]

~~6.7.A.4 Annual Radioactive Effluent Report~~

~~Routine radioactive effluent reports covering the operation of the unit during the previous calendar year of operation shall be submitted by May 1st of each year.~~

~~The radioactive effluent report shall include a summary of the quantities of radioactive liquid and gaseous effluents as outlined in Appendix B of Regulatory Guide 1.21, Revision 1, June, 1974, with data summarized on a quarterly basis.~~

~~The report shall include an assessment of the radiation doses form radioactive effluents released from the plant during the previous calendar year. The report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to individuals due to their activities inside the site boundary (Figures 3.9-1 and 3.9-2) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the report. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM) or standard NRC computer codes.~~

~~The report shall also include an assessment of radiation doses to the likely most exposed member of the general public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation.~~

~~The radioactive effluent report shall include the following information for solid waste shipped offsite during the report period.~~

~~a. container volume,~~
~~b. total curie quantity (specify whether determined by measurement or estimate),~~
~~c. principal radionuclides (specify whether determined by measurement or estimate),~~
~~d. type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),~~
~~e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and~~
~~f. solidification agent (e.g., cement, urea formaldehyde).~~

~~The radioactive effluent report shall include unplanned releases from the site of radioactive materials in gaseous and liquid effluents on a quarterly basis, changes to the ODCM, a description of changes to the PCP, a report of when milk or vegetable samples cannot be obtained as required by Table 4.10-1, and changes in land use resulting in significant increases in calculated doses.~~

D.~~3.~~Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.
~~A monthly report of operating statistics and shutdown experience covering the previous month shall be submitted by the 15th of the following month to the Director of the Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555.~~

~~6.7.A.5. Annual Summaries of Meteorological Data~~

~~An annual summary of meteorological data shall be submitted for the previous calendar year in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability at the request of the Commission.~~

~~6.7.A.6.~~E. Core Operating Limits Report (COLR)

1~~a~~. Core operating limits shall be established prior to ~~and documented in the CORE OPERATING LIMITS REPORT before~~ each reload cycle, or prior to any remaining portion ~~part~~ of a reload cycle, and shall be documented in the COLR for the following:

a~~1~~. Heat Flux Hot Channel Factor Limit ($F_Q^{RPT}$), Nuclear Enthalpy Rise Hot Channel Factor Limit ($F\Delta_H^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

b~~2~~. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

c~~3~~. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

d~~4~~. Reactor Coolant System Flow Limit (Specification 3.10.J)

2~~b~~. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

NSPNAD-8102-A, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version)

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990

XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981

WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993).

NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version)

3~~c~~. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, ~~and~~ transient analysis limits and accident analysis limits) of the safety analysis are met.

4~~d~~. The C~~ORE OPERATING LIMITS REPORT~~, including any mid-cycle revisions or supplements ~~thereto~~, shall be provided ~~supplied~~ upon issuance~~,~~ for each reload cycle~~,~~ to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector~~.

~~B. REPORTABLE EVENTS~~

~~The following actions shall be taken for REPORTABLE EVENTS:~~

~~a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and~~

~~b. Each REPORTABLE EVENT shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the Vice President Nuclear Generation.~~

~~C. Environmental Reports~~

~~The reports listed below shall be submitted to the Administrator of the appropriate Regional NRC Office or his designate:~~

[Paragraphs 6.7.C.1 and 2 relocated above]

~~3. Other Environmental Reports (non-radiological, non-aquatic)~~

~~Written reports for the following items shall be submitted to the appropriate NRC Regional Administrator:~~

~~a. Environmental events that indicate or could result in a significant environmental impact casually related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by the Endangered Species Act of 1973; or increase in nuisance organisms or conditions. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant~~

~~operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.~~

~~b. Proposed changes, test or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.~~

~~D. Special Reports~~

~~Unless otherwise indicated, special reports required by the Technical Specifications shall be submitted to the appropriate NRC Regional Administrator within the time period specified for each report.~~

## 6.7 High Radiation Area

[Revision of paragraph 6.5.B.1 was proposed in License Amendment Request, "Radiological Effluent Technical Specifications Conformance to Standard Technical Specifications and Generic Letter 89-01", Dated July 17, 1995. This submittal proposes to move these requirements to a new Specification Section 6.7 entitled, "High Radiation Area"]

~~3.1 REACTOR COOLANT SYSTEM~~

~~Bases continued~~

~~E. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration~~

~~By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the normal steady-state operation limits specified, the integrity of the reactor coolant system is assured under all operating conditions (Reference 1).~~

~~If these steady-state limits are exceeded, measures can be taken to correct the condition during reactor operation, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank (Reference 2). Because of the time dependent nature of any adverse effects from oxygen, chloride, and fluoride concentrations in excess of the limits, it is unnecessary to shut down immediately since the conditions for corrective action to restore concentrations within the steady-state limits has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the COLD SHUTDOWN condition and the corrective action will continue.~~

~~The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit transient concentrations to exist for 48 hours for coolant temperatures less than 250°F and still provide the assurance the integrity of the primary coolant system will be maintained.~~

~~In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.~~

~~References~~

~~1. USAR, Section 4.5.2~~
~~2. USAR, Section 10.2.3~~

## 4.4 CONTAINMENT SYSTEM TESTS

**Bases** continued

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the Shield Building Ventilation System (Reference 5). Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the Auxiliary Building Special Ventilation System. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of-coolant accident was initially evaluated by the AEC staff (Reference 3) assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest SITE BOUNDARY and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment in-leakage (Reference 6).

~~The Residual Heat Removal Systems functionally become a part of the containment volume during the post-accident period when their operation is changed over from the injection phase to the recirculation phase. Redundancy and independence of the systems permit a leaking system to be isolated from the containment during this period, and the possible consequences of leakage are minor relative to those of the Design Basis Accident (Reference 4); however, their partial role in containment warrants surveillance of their leak-tightness.~~

~~The limiting leakage rates from the recirculation heat removal system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure, 350 psig, gives an adequate margin over the highest pressure within the system after a design basis accident. A recirculation heat removal system leakage of 2 gal/hr will limit off-site exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.~~

LICENSE AMENDMENT REQUEST DATED December 14, 1995

Conformance of Administrative Controls Section 6
to the Guidance of Standard Technical Specifications

EXHIBIT C

Appendix A, Technical Specification Pages
Revised Pages

TS-ii
TS-v
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# TECHNICAL SPECIFICATIONS

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## APPENDIX A TECHNICAL SPECIFICATIONS

## LIST OF FIGURES

3.1.D. MAXIMUM COOLANT ACTIVITY

1. The specific activity of the primary coolant (except as specified in 3.1.D.2 and 3 below) shall be limited to:

   a. Less than or equal to 1.0 microcuries per gram DOSE EQUIVALENT I-131, and

   b. Less than or equal to $100/\bar{E}$ microcuries per gram of gross radioactivity.

2. If a reactor is critical or the reactor coolant system average temperature is greater than or equal to 500°F:

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure TS.3.1-3, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

   b. With the specific activity of the primary coolant greater than 100/E microcurie per gram, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

3. If a reactor is at or above COLD SHUTDOWN, with the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/E microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.1-2B until the specific activity of the primary coolant is restored to within its limits.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

| TEST | FREQUENCY |
|---|---|
| 1. RCS Gross Activity Determination | 5/week |
| 2. RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | 1/14 days (when at power) |
| 3. RCS Radiochemistry $\bar{E}$ determination | 1/6 months(1) (when at power) |
| 4. RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135 | a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and<br>b) One sample between 2 and 6 hours following thermal POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period ( above hot shutdown) |
| 5. RCS Radiochemistry (2) | Monthly |
| 6. RCS Tritium Activity | Weekly |
| 7. Deleted | |
| 8. RCS Boron Concentration*(3) | 2/Week (4) |
| 9. RWST Boron Concentration | Weekly |
| 10. Boric Acid Tanks Boron Concentration | 2/Week |
| 11. Caustic Standpipe NaOH Concentration | Monthly |
| 12. Accumulator Boron Concentration | Monthly |
| 13. Spent Fuel Pit Boron Concentration | Monthly/Weekly(7)(8) |

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.

C. Containment Vacuum Breakers

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containmant vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested during each refueling shutdown in accordance with the requirements of Specification 4.4.A.2.

## 4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

### Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

### Objective

To verify that the emergency power sources and equipment are OPERABLE.

### Specification

The following tests and surveillance shall be performed:

A. Diesel Generators

1. At least once each month, for each diesel generator:

    a. Verify the fuel level in the day tank.

    b. Verify the fuel level in the fuel storage tank.

    c. Deleted

    d. Verify the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

    e. Verify the diesel generator can start and gradually accelerate. Verify the generator voltage and frequency can be adjusted to 4160 ± 420 volts and 60 ± 1.2 Hz. Subsequently, manually sychronize the generator, gradually load to at least 1650 kW (Unit 2: 5100 kW to 5300 KW), and operate for at least 60 minutes. This test should be conducted in consideration of the manufacturer's recommendations regarding engine prelube, warm-up, loading and shutdown procedures where possible.

# 5.0 DESIGN FEATURES

## 5.1 SITE

The Prairie Island Nuclear Generating Plant is located on property owned by Northern States Power (NSP) Company at a site on the west bank of the Mississippi River, approximately 6 miles northwest of the city of Red Wing, Minnesota. The minimum distance from the center line of either reactor to the site exclusion boundary is 715 meters, and the low population zone distance is 1-1/2 miles. The nearest population center of 25,000 or more people is South Saint Paul. These site characteristics comply with definitions in 10CFR100 (Reference 1).

The U.S. Army Corp of Engineers controls the land within the exclusion area that is not owned by NSP. The Corps has made an agreement with NSP to prevent residential construction on this land for the life of the plant (Reference 2).

These specifications use atmospheric diffusion factors based on the NRC staff evaluations. Its evaluation of accidental airborne releases is based on a relative concentration of $9.8 \times 10^{-4}$ seconds per cubic meter at the site boundary. Its evaluation of routine releases is based on a relative concentration of $1.5 \times 10^{-5}$ seconds per cubic meter (Reference 3).

The flood of record in 1965 produced a water surface elevation of +688 feet MSL at the site. The calculated probable maximum flood (PMF) level is +703.6 feet mean sea level (MSL), and the estimated wave runup could reach +706.7 feet MSL. (See Section 2.4.2 of this report.) Plant grade level is +695 feet MSL.

Flood protection structures have been provided. The two turbine support facilities, the common auxiliary building, and the two shield buildings have been physically connected by a concrete flood wall, most of the length of which constitutes the concrete foundation walls for the various buildings. The top of this wall supports the metal siding for the buildings at about elevation +705 feet MSL. Fourteen doors through the flood wall, or into the various buildings (including the separate screen house), are provided with receivers for the erection of flood protection panels to prevent flood water from reaching safety related facilities.

The cooling water pumps in the screenhouse are designed to operate up to a flood level of +695 feet MSL without flood protection measures, and up to a level of +707 feet MSL with the erection of flood protection panels. The main transformer foundation is at +695 feet MSL. The transformer will function to a flood level of+698 feet MSL.

The plant is designed for a design basis earthquake having a horizontal ground acceleration of 0.12g and an operational basis earthquake having a horizontal ground acceleration of 0.06g.

References

1. USAR, Section 2.2.1
2. USAR, Section 3.4.5
3. SER, Sections 2.3.4 and 2.3.5

# 6.0 ADMINISTRATIVE CONTROLS

## 6.1 Responsibility

A. The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

B. The shift supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active senior reactor operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator license shall be designated to assume the control room command function.

## 6.2 Organization

### A. Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

1. Lines of authority, responsibility and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Safety Analysis Report.

2. The plant manager shall report to the corporate vice president specified in 6.2.A.3, shall be responsible for overall safe operation of the plant, and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

3. A corporate vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

### B. Plant Staff

The plant staff organization shall include the following:

1. An operator to perform non-licensed duties shall be assigned to each reactor containing fuel and one additional operator to perform non-licensed duties shall be assigned when either or both reactors are operating in MODES 1, 2, 3, or 4. Also, if one unit is in MODE 1, 2, 3, or 4 and the other unit is in MODE 5 or 6, as a minimum the on-site staffing shall include two senior reactor operators(SRO) and two licensed reactor operators(RO).

2. At least one licensed operator shall be present in the control room for each reactor containing fuel. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed senior reactor operator shall be present in the control room

3. Shift crew composition may be less than the minimum requirement of 10CFR50.54(m)(2)(i) and 6.2.B.1 and 6.2.B.7 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. The amount of overtime worked by plant staff members performing safety related functions shall be limited and controlled by procedures which implement an NRC approved program.

6. The operations manager or assistant operations manager shall hold an SRO license.

7. The shift technical advisor (STA) shall provide advisory technical support to the shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. Personnel performing the function of the STA shall be assigned to the shift crew when a unit is in MODE 1, 2, 3, or 4.

## 6.3 Plant Staff Qualifications

Each member of the plant staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975 except for (1) personnel who perform the function of shift technical advisor shall have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (2) the operations manager who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.2.B.6.

6.4 Procedures

Written procedures shall be established, implemented, and maintained covering the following activities:

A. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

B. The emergency operating procedures required to implement the requirements of NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

C. Quality control for effluent and environmental monitoring;

D. Fire protection program implementation; and

E. All programs specified in Specification 6.5.

## 6.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

A. Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Monitoring, and Radioactive Effluent Reports required by Specification 6.6.B and Specification 6.6.C.

Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   b. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations;

2. Shall become effective after approval by a member of plant management designated by the Plant Manager.

3. Shall be submitted to the NRC in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed. The date (i.e., month and year) the change was implemented shall be indicated.

B. Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The systems include portions of Residual Heat Removal, Safety Injection, and Containment Spray Systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

C. Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel;

2. Procedures for sampling and analysis; and

3. Provisions for maintenance of sampling and analysis equipment.

D. Radioactive Effluent Controls Program

This program conforms to 10CFR50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable.

This program shall allocate releases equally to each unit. The liquid radwaste treatment system, waste gas treatment system, containment purge release vent, and spent fuel pool vent are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. Therefore, all releases will be allocated equally in determining conformance to the design objectives of 10CFR50, Appendix I.

The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

2. Limitation on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to Appendix B to 10CFR20.1 - 20.601, Table II, Column 2;

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10CFR20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10CFR50, Appendix I;

5. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least monthly;

6. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate

portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of one month from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with Appendix B to 10CFR20.1 - 20.601, Table II, Column 1;

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10CFR50, Appendix I;

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10CFR50, Appendix I; and

10. Limitation on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40CFR190.

E. <u>Component Cyclic or Transient Limit</u>

This program provides controls to track the USAR, Section 4.1.4 cyclic and transient occurrences to ensure that components are maintained within the design limits.

F. (Reserved)

G. (Reserved)

H. (Reserved)

I. (Reserved)

J. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentration of oxygen in the waste gas holdup system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria;

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 78,000 curies of noble gases (considered as dose equivalent Xe-133); and

3. A surveillance program to ensure that the quantity of radioactivity contained in each of the following tanks shall be limited to 10 curies, excluding tritium and dissolved or entrained noble gases:

   Condensate storage tanks
   Outside temporary tanks

4. The provisions of TS 4.0 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

K. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment.

L. Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases or the Technical Specifications shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   a. a change in the Technical Specifications incorporated in the license; or

   b. a change to the USAR or Bases that involves an unreviewed safety question as defined in 10CFR50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

4. Proposed changes that meet the criteria of Specification 6.5.L.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with USAR updates.

## 6.6 Reporting Requirements

The following reports shall be submitted in accordance with 10CFR50.4

### A. Occupational Exposure Report

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10CFR20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. This report shall be submitted by April 30 of each year.

### B. Annual Radiological Environmental Monitoring Report

The Annual Radiological Environmental Monitoring Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10CFR50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiation Environmental Monitoring Reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

### C. Radioactive Effluent Report

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR50.36a and 10CFR50, Appendix I, Section IV.B.1.

D. Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

E. Core Operating Limits Report (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   a. Heat Flux Hot Channel Factor Limit ($F_Q^{RPT}$), Nuclear Enthalpy Rise Hot Channel Factor Limit ($F\Delta_H^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

   b. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

   c. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

   d. Reactor Coolant System Flow Limit (Specification 3.10.J)

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

   NSPNAD-8102-A, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version)

   WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

   WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985

   WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

   WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990

   XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981

   WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993).

NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version)

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

4. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

## 6.7 High Radiation Area

A. Pursuant to 10CFR20, paragraph 20.1601(c), in lieu of the requirements of 10CFR20.1601, each high radiation area, as defined in 10CFR20, in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., health physics technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates less than or equal to 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

   Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   1. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

   2. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

   3. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection manager.

B. In addition to the requirements of Specification 6.7.A above, areas with radiation levels greater than or equal to 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV or transmitting radiation monitoring device) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

C. For individual high radiation areas with radiation levels of greater than 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

## 4.4 CONTAINMENT SYSTEM TESTS

Bases continued

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the Shield Building Ventilation System (Reference 5). Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the Auxiliary Building Special Ventilation System. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of-coolant accident was initially evaluated by the AEC staff (Reference 3) assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest SITE BOUNDARY and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment in-leakage (Reference 6).

LICENSE AMENDMENT REQUEST DATED December 14, 1995

Conformance of Administrative Controls Section 6
<u>To the Guidance of Standard Technical Specifications</u>

EXHIBIT D

Standard Technical Specification Pages
Marked Up Pages

LICENSE AMENDMENT REQUEST DATED November 2, 1995

Conformance of Administrative Controls Section 6
<u>To the Guidance of Standard Technical Specifications</u>

EXHIBIT D

Standard Technical Specification Pages
Marked Up Pages

# 6 ~~5~~.0 ADMINISTRATIVE CONTROLS

## 6 ~~5~~.1 Responsibility

6.1 ~~5.1.1~~ A. The ~~[Plant Superintendent]~~ manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The ~~[Plant Superintendent]~~ manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

~~5.1.2~~ B. The ~~[~~Shift Supervisor (SS)~~]~~ shall be responsible for the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

~~5.0 ADMINISTRATIVE CONTROLS~~

## 6 ~~5~~.2 Organization

### ~~5.2.1~~ A Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for ~~unit~~ plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

1. ~~a.~~ Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements Insert 1 shall be documented in the ~~[FSAR].~~ Updated Safety Analysis Report.

2. ~~b.~~ The ~~[P~~lant ~~Superintendent]~~ manager shall report to the corporate Vice President, specified in 6.2.A.3, shall be responsible for overall safe operation of the plant, and shall have control over those onsite activities necessary for safe operation and maintenance of the plant~~;~~.

3. ~~c.~~ ~~The [a specified~~ A corporate ~~executive position]~~ Vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety~~; and~~.

4. ~~d.~~ The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

### ~~5.2.2~~ B ~~Unit~~ Plant Staff

The ~~unit~~ plant staff organization shall include the following:

1. ~~a.~~ An operator to perform ~~A~~ non-licensed ~~operator~~ duties shall be assigned to each reactor containing fuel and ~~an~~ one additional operator to perform non-licensed ~~operator~~ duties

~~(continued)~~

Insert 1, STS page 5.0-2

. . ., including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specification, . . .

Exhibit D
Standard Technical Specifications
Marked Up Pages

## 6 ~~5~~.2 Organization

~~5.2.2~~ ~~Unit Staff (continued)~~

when either or both

shall be assigned ~~for each control room from which a~~ reactors are ~~is~~ operating in MODES 1, 2, 3, or 4. Insert 2

~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~

2. ~~b~~. At least one licensed ~~Reactor~~ Operator ~~(RO)~~ shall for each be present in the control room ~~when fuel is in the~~ reactor containing fuel. In addition, while ~~the~~ unit is either in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator ~~(SRO)~~ shall be present in the control room.

3 ~~c~~. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and ~~5.2.2.a~~ 6.2.B.1 and ~~5.2.2.g~~ 6.2.B.7 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

4 ~~d~~. ~~A [Health Physics Technician]~~ An individual qualified in radiation protection procedures shall be on site when fuel is in a ~~the~~ reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

~~e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).~~

~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

~~1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;~~

~~(continued)~~

Insert 2, STS page 5.0-3

Also, if one unit is in MODE 1, 2, 3, or 4 and the other unit is in MODE 5 or 6, as a minimum the on-site staffing shall include two senior reactor operators (SRO) and two licensed reactor operators(RO).

Exhibit D
Standard Technical Specifications
Marked Up Pages

6 ~~5~~.2 Organization

~~5.2.2~~ ~~Unit Staff (continued)~~

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

~~OR~~

5. [ The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled ~~in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12)~~ by procedures which implement an NRC approved program. ]

6. ~~f~~. The ~~[~~Operations Manager or Assistant Operations Manager~~]~~ shall hold an SRO license.

7. ~~g~~. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor ~~(SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~ Insert 3

Insert 3, STS page 5.0-4

Personnel performing the function of the STA shall be assigned to the shift crew when a unit is in MODE 1, 2, 3, or 4.

Exhibit D
Standard Technical Specifications
Marked Up Pages

~~5.0 ADMINISTRATIVE CONTROLS~~

## 6 ~~5~~.3 ~~Unit~~ Plant Staff Qualifications

~~Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[~~Regulatory Guide 1.8, Revision ~~2~~ 1, September 1975, ~~1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ Insert 4

Insert 4, STS page 5.0-5

. . . except for (1) personnel who perform the function of shift technical advisor shall have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (2) the operations manager who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.2.B.6.

Exhibit D
Standard Technical Specifications
Marked Up Pages

~~5.0 ADMINISTRATIVE CONTROLS~~

## 6~~5~~.4 Procedures

~~5.4.1~~ Written procedures shall be established, implemented, and maintained covering the following activities:

A ~~a.~~ The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

B ~~b.~~ The emergency operating procedures required to implement the requirements of NUREG-0737 ~~and to NUREG-0737,~~ Supplement 1, as stated in ~~[~~Generic Letter 82-33~~]~~;

C ~~c.~~ Quality control ~~assurance~~ for effluent and environmental monitoring;

D ~~d.~~ Fire ~~P~~rotection ~~P~~rogram implementation; and

E ~~e.~~ All programs specified in Specification 6~~5~~.5.

~~5.0 ADMINISTRATIVE CONTROLS~~

## 6 ~~5~~.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

### ~~5.5.1~~ A Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.6.2] and Specification [5.6.3].

Licensee initiated changes to the ODCM:

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the [Plant Superintendent]; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)~~

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

## ~~5.5.2~~ B. Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~[Recirculation Spray~~ portions of Residual Heat Removal, Safety Injection, ~~Chemical and Volume Control, gas stripper, and Hydrogen Recombiner]~~ and Containment Spray Systems. The program shall include the following:

1 ~~a.~~ Preventive maintenance and periodic visual inspection requirements; and

2 ~~b.~~ Integrated leak test requirements for each system at refueling cycle intervals or less.

## ~~5.5.3~~ C. Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

1 ~~a.~~ Training of personnel;

2 ~~b.~~ Procedures for sampling and analysis; and

3 ~~c.~~ Provisions for maintenance of sampling and analysis equipment.

## ~~5.5.4~~ D. Radioactive Effluent Controls Program

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to

~~(continued)~~

~~5.5.4~~ Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.4~~ ~~Radioactive Effluent Controls Program~~ (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

E. ~~5.5.5~~ Component Cyclic or Transient Limit

This program provides controls to track the U ~~FSAR~~, Section 4.1.4 ~~[ ]~~, cyclic and transient occurrences to ensure that components are maintained within the design limits.

~~5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.~~

F. (Reserved))

~~5.5.7 Reactor Coolant Pump Flywheel Inspection Program~~

~~This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~

G. (Reserved)

~~(continued)~~

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

H. (Reserved)

~~5.5.9 Steam Generator (SG) Tube Surveillance Program~~

[Reviewer's Note: The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.]

~~(continued)~~

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

F. Reserved

~~5.5.11 Ventilation Filter Testing Program (VFTP)~~

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in [Regulatory Guide ], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1].

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

| ESF Ventilation System | Flowrate |
|---|---|
| [ ] | [ ] |

~~(continued)~~

5.5 Programs and Manuals

5.6.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

| ESF Ventilation System | Flowrate |
|---|---|
| [ ] | [ ] |

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of ≤ [30°C] and greater than or equal to the relative humidity specified below.

| ESF Ventilation System | Penetration | RH |
|---|---|---|
| [ ] | [ ] | [ ] |

[ Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).

Safety factor = [5] for systems with heaters.
= [7] for systems without heaters. ]

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52,

~~(continued)~~

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

| ESF Ventilation System | Delta P | Flowrate |
|---|---|---|
| [ ] | [ ] | [ ] |

e. [Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].]

| ESF Ventilation System | Wattage |
|---|---|
| [ ] | [ ] |

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

~~5.5.12~~ J. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the [Waste Gas Holdup System] and a surveillance program to ensure the limits are maintained. Such limits shall be

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.12~~ Explosive Gas and Storage Tank Radioactivity Monitoring Program ~~(continued)~~

appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents]: and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

~~5.5.13~~ k. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with ~~applicable~~ ASTM D975-77 ~~Standards. The purpose of the program is to establish the following:~~

the limits specified in Table 1 of

when checked for viscosity, water, and sediment.

~~a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:~~

~~1. an API gravity or an absolute specific gravity within limits,~~

~~(continued)~~

~~5.5.13 Diesel Fuel Oil Testing Program (continued)~~

~~2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and~~

~~3. a clear and bright appearance with proper color;~~

~~b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and~~

~~c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.~~

~~5.5.14~~ L. Technical Specifications ~~(TS)~~ Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. ~~a.~~ Changes to the Bases of the ~~TS~~ Technical Specifications shall be made under appropriate administrative controls and reviews.

2. ~~b.~~ Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   a. ~~1.~~ a change in the ~~TS~~ Technical Specifications incorporated in the license; or

   b. ~~2.~~ a change to the ~~updated F~~USAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

3. ~~c.~~ The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the ~~F~~USAR.

4 ~~d.~~ Proposed changes that meet the criteria of Specification ~~5.5.14b~~ above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with ~~10 CFR 50.71(e).~~ USAR updates.

6.5.L.2

~~5.5 Programs and Manuals (continued)~~

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

~~5.0 ADMINISTRATIVE CONTROLS~~

## 6 ~~5~~.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

### ~~5.6.1~~ A. Occupational Radiation Exposure Report

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than ~~>~~ 100 mrem/yr and their associated man rem exposure according to work and job functions, ~~(~~e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (~~[~~describe maintenance~~]~~) waste processing, and refueling~~)~~. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter ~~(TLD)~~, or film badge measurements. Small exposures totalling ~~<~~ less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. ~~[The initial report shall be submitted by April 30 of the year following the initial criticality.]~~

### ~~5.6.2~~ B. Annual Radiological Environmental Operating Report

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.2~~ ~~Annual Radiological Environmental Operating Report~~ (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

~~5.6.3~~ C. Radioactive Effluent ~~Release~~ Report

-----------------------------NOTE-----------------------------
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~
--------------------------------------------------------------

The Radioactive Effluent ~~Release~~ Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

~~(continued)~~

~~5.6 Reporting Requirements (continued)~~

~~5.6.4~~ D. Monthly Operating Reports

Routine reports of operating statistics and shutdown experience~~[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves.]~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

~~5.6.5~~ E. ~~CORE OPERATING LIMITS REPORT~~ Core Operating Limits Report (COLR)

1. ~~a.~~ Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: Insert 5

~~[The individual specifications that address core operating limits must be referenced here.]~~

2. ~~b.~~ The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: Insert 6

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~

3. ~~c.~~ The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ~~Emergency Core Cooling Systems (ECCS)~~ limits, nuclear limits such as ~~SDM~~ Shut Down Margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

4. ~~d.~~ The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic~~

~~(continued)~~

Insert 5, STS page 5.0-20

a. Heat Flux Hot Channel Factor Limit ($F_Q^{RPT}$), Nuclear Enthalpy Rise Hot Channel Factor Limit ($F\Delta_H^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

b. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

c. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

d. Reactor Coolant System Flow Limit (Specification 3.10.J)

Insert 6, STS page 5.0-20

NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

NSPNAD-8102-A, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version)

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990

XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981

WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993).

NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version)

Exhibit D
Standard Technical Specifications
Marked Up Pages

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

~~(continued)~~

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

[6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature ($RT_{NDT}$) to the predicted increase in $RT_{NDT}$; where the predicted increase in $RT_{NDT}$ is based on the mean shift in $RT_{NDT}$ plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{NDT}$ + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.]

5.6.7 EDG Failure Report

[If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.[3], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

~~(continued)~~

5.6 Reporting Requirements (continued)

5.6.9 [Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]

5.6.10 [Steam Generator Tube Inspector Report

Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.]

[Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.]

T.S.6.0-13

~~5.0 ADMINISTRATIVE CONTROLS~~

6 ~~[5~~.7 High Radiation Area~~]~~

~~5.7.1~~ Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

This section is not marked up since it was previously submitted for review by the NRC on July 17, 1995

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in

~~(continued)~~