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ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

MASTER RELAY TEST

1.17 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC means an individual in a controlled area or UNRESTRICTED AREA. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

1.19 Not Used

OPERABLE - OPERABILITY

1.20 A system, subsystem, train, component or device shall be OPERABLE or have OPERABLITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.21 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.22 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.23 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

1.24 Not Used

PURGE - PURGING

1.25 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.26 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3,853 Mwt (Model Δ94 steam generators installed) or 3,800 Mwt (Model E steam generators installed).

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components or methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in the Core Operating Limits Report.

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlation and $\geq$ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58 °F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of 10 CFR 50.36(c)(1).

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, AND 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of 10 CFR 50.36(c)(1).

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of 10 CFR 50.36(c)(1).

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Core Operating Limits Report (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-2a.

APPLICABILITY: Beginning of Life (BOL) limit - MODES 1 and 2* only**.
End of Life (EOL) limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

   3. A Special Report is submitted within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels, transfer switches, power or control circuits less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown monitoring channels, transfer switches, power or control circuits less than the Total Number of Channels as required by Table 3.3-9, within 60 days restore the inoperable channel(s) to OPERABLE status or, submit a Special Report that defines the corrective action to be taken.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at least once per 18 months.

ACTION STATEMENTS (Continued)

ACTION 39 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 40 - With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 41 - a. With the number of OPERABLE channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or submit a Special Report within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Submit a Special Report within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the system to OPERABLE status at the next scheduled refueling.

ACTION 42 - a. With one required channel inoperable, restore the required channel to OPERABLE status within 30 days; otherwise, a Special Report shall be submitted within the next 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 to OPERABLE status.

b. With two required channels inoperable, restore one required channel to OPERABLE status within 7 days; otherwise, be in HOT STANDBY within 6 hours, and in HOT SHUTDOWN in the next 6 hours.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions [plug or (for Model E steam generators only) repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks] required by Table 4.4-2 and Table 4.4-3.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported in a Special Report;

b. The complete results of the steam generator tube inservice inspection shall be submitted in a Special Report within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Model E steam generators, implementation of the voltage-based repair criteria to tube support plate intersections, notify the Staff prior to returning the steam generators to service should any of the following conditions arise:

1) If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle. The calculation shall be done using:

a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot-leg support plates L through R; and

LIMITING CONDITION FOR OPERATION

3.4.9.3 An Overpressure Protection System shall be OPERABLE with a maximum of one centrifugal charging pump capable of injecting into the RCS and the emergency core cooling system (ECCS) accumulators isolated and either a. or b. below:

a. Two power-operated relief valves (PORVs) with lift settings which do not exceed the limit established in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.0 square inches.

APPLICABILITY: MODES 4 and 5, and MODE 6 when the head is on the reactor vessel[1].

ACTION:

a. With one or more ECCS accumulators not isolated, isolate the ECCS accumulator(s) within 1 hour.

b. With more than one centrifugal charging pump capable of injecting into the RCS, immediately initiate action to render all but one centrifugal charging pump incapable of injecting into the RCS[2].

c. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.0 square inch vent within the next 8 hours.

d. With one PORV inoperable in MODES 5 or 6 with the head on the reactor vessel, restore the inoperable PORV to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 2.0 square inch vent within the next 8 hours[3].

e. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.0 square inch vent within 8 hours[3].

f. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be submitted within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

g. The provisions of Specification 3.0.4 are not applicable.

3/4.5.2 ECCS SUBSYSTEMS - T_{AVG} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Three independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE High Head Safety Injection pump,

b. One OPERABLE Low Head Safety Injection pump,

c. One OPERABLE RHR heat exchanger, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation through a High Head Safety Injection pump and into the Reactor Coolant System and through a Low Head Safety Injection pump and its respective RHR heat exchanger into the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With less than the above subsystems OPERABLE, but with at least two High Head Safety Injection pumps in an OPERABLE status, two Low Head Safety Injection pumps and associated RHR heat exchangers in an OPERABLE status, and sufficient flow paths to accommodate these OPERABLE Safety Injection pumps and RHR heat exchangers, ** restore the inoperable subsystem(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be submitted within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.1.2 provided that the Safety Injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

**Verify required pumps, heat exchangers and flow paths OPERABLE every 48 hours.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3.1 As a minimum, the following ECCS components shall be OPERABLE:

a. Two OPERABLE High Head Safety Injection pumps,*

b. Two OPERABLE Low Head Safety Injection pumps and their associated RHR heat exchangers, and

c. Two OPERABLE flow paths capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation through a High Head Safety Injection pump and into the Reactor Coolant System and through a Low Head Safety Injection pump and its respective RHR heat exchanger into the Reactor Coolant System.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above-required ECCS components OPERABLE because of the inoperability of either the High Head Safety Injection pumps or the flow paths from the refueling water storage tank, restore at least the required ECCS components to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With less than the above-required ECCS components OPERABLE because of the inoperability of either the residual heat removal heat exchangers or the Low Head Safety Injection pumps, restore at least the required ECCS components to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be submitted within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*A maximum of one High Head Safety Injection pump shall be OPERABLE and a second High Head Safety Injection pump shall be OPERABLE except that its breaker shall be racked out (the third HHSI pump shall have its breaker racked out) within: (1) 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first; or (2) 4 hours after entering MODE 4 from MODE 5 or prior to the temperature of one or more of the RCS cold legs exceeding 225°F, whichever comes first.

LIMITING CONDITION FOR OPERATION

ACTION (continued)

g. With one or more diesel generator fuel oil storage tanks with stored fuel oil total particulates not within the Diesel Fuel Oil Testing Program limits, within 7 days restore the fuel oil total particulates within limits, or declare the associated standby diesel generator(s) inoperable.

h. With one or more diesel generator fuel oil storage tanks with new fuel oil properties not within the Diesel Fuel Oil Testing Program limits, within 30 days restore the fuel oil properties within limits, or declare the associated standby diesel generator(s) inoperable.

6.1.1 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 the plant manager's designee shall approve, prior to implementation, each proposed test and experiment that affects nuclear safety and is not described in the UFSAR, and each modification to systems or equipment that affects nuclear safety.

6.1.2 The shift manager shall be responsible for the control room command function. During any absence of the shift manager 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 for that unit. During any absence of the shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function for that unit.

6.2.1 Offsite and Onsite Organizations

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

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, 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 UFSAR and/or the Operations Quality Assurance Plan.

b. The plant manager 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.

c. A specified corporate officer 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.

d. The individuals who train the operating staff, carry out radiation protection functions, 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.

6.2.2 Unit Staff

The unit staff organization shall include the following:

a. A total of three non-licensed operators for the two units is required in all conditions. At least one of the required non-licensed operators shall be assigned to each unit. When a unit is operating in MODES 1, 2, 3, or 4, two non-licensed operators are required to be assigned to that unit.

b. The shift crew composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.2.a and 6.2.2.g 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.

(continued)

6.2.2 <u>Unit Staff</u> (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

d. Administrative controls shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, reactor plant operators, and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from these guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, with documentation of the basis for granting the deviation.

Controls shall be included in the procedures to require that a periodic independent review be conducted to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

e. The individual to whom the shift managers directly report shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift manager 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 (Generic Letter 86-04). This position may also be filled by the shift manager or an individual with an SRO license provided that person meets the qualifications specified by the Commission Policy Statement.

6.2.3 <u>Not Used</u>

6.2.4 <u>Not Used</u>

6.0 ADMINISTRATIVE CONTROLS
6.3 Unit Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, as described in the Operations Quality Assurance Plan.

6.4 Not Used

6.5 Not Used

6.6 Not Used

6.7 Not Used

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Quality Assurance Program for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. Programs and Manuals specified in Specification 6.8.3.

6.8.2 Not Used

6.8.3 The following programs and manuals shall be established, implemented, and maintained:

a. 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 containment spray, safety injection, containment hydrogen monitoring, and primary sampling. 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.

b. Not Used

c. Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

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

2) Identification of the procedures used to measure the values of the critical variables;

(continued)

6.8.3.c (continued)

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

4) Procedures for the recording and management of data;

5) Procedures defining corrective actions for all off-control point chemistry conditions; and

6) A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

d. Not used

e. Not Used

f. Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1 cyclic/transient plant conditions to assure that the components are maintained within the design limits.

g. Radioactive Effluent Controls Program

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 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;

(continued)

6.8.3.g (continued)

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

3) 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;

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY from the radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to 10 CFR 50, 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 every 31 days;

6) Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that the 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 percent of the guidelines for the annual dose or dose commitment conforming to 10 CFR 50, Appendix I;

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the following:

   a) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas at or beyond the SITE BOUNDARY conforming to 10 CFR 50, 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 8 days in gaseous effluents released from each unit to areas at or beyond the SITE BOUNDARY conforming to 10 CFR 50, Appendix I; and

(continued)

6.8.3.g (continued)

10) 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.

h. <u>Not Used</u>

i. <u>Diesel Fuel Oil Testing Program</u>

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 based on applicable ASTM Standards. The purpose of the program is to establish the following:

1) Acceptability of new fuel oil prior to addition to the diesel generator fuel oil storage tanks by determining that the fuel oil has:

   a. an API gravity or absolute specific gravity within limits,

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

   c. a clear and bright appearance with proper color;

2) Within 31 days following addition of new fuel oil to the diesel generator fuel oil storage tanks, verify that the properties of the new fuel oil, other than those addressed in 6.8.3.i.1 above, are within limits for ASTM 2D fuel oil; and

3) Total particulate concentration of fuel oil is ≤ 10 mg/l when tested every 31 days using a test method based on ASTM D-2276.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

j. <u>Containment Leakage Rate Testing Program</u>

A program shall be established to implement leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995. The current ten-year interval between performance of the integrated leakage rate (Type A) test, beginning September 24, 1991, for Unit 2 and March 10, 1995, for Unit 1, has been extended to 15 years (a one-time change).

(continued)

6.8.3.j (continued)

Peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA), P_a is 41.2 psig.

The maximum allowable containment leakage rate, L_a, is 0.3 percent of containment air weight per day.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit start-up following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ as-left and $\leq 1.0\ L_a$ as-found for Type A tests.

2) Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Surveillance Requirement 4.0.3 apply to the Containment Leakage Rate Testing Program.

k. <u>Configuration Risk Management Program (CRMP)</u>

A program to assess changes in core damage frequency and cumulative core damage probability resulting from applicable plant configurations. The program should include the following:

1) training of personnel;

2). procedures for identifying plant configurations, the generation of risk profiles and the evaluation of risk against established thresholds; and

3) provisions for evaluating changes in risk resulting from unplanned maintenance activities.

(continued)

6.8.3 (continued)

l. <u>Containment Post-Tensioning System Surveillance Program</u>

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 measures prior to initial operations. The Containment Post-Tensioning System Surveillance Program shall be in accordance with ASME Code Section XI, Subsection IWL, 1992 Edition with 1992 Addenda, as supplemented by 10CFR50.55a(b)(2)(viii).

m. <u>Technical Specifications (TS) Bases Control Program</u>

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

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

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

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of Specification 6.8.3.m.b 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).

n. <u>Offsite Dose Calculation Manual (ODCM)</u>

1) 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

(continued)

6.8.3.n (continued)

2) The ODCM shall also contain descriptions of the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radiological Effluent Release Report required by Specifications 6.9.1.3 and 6.9.1.4.

3) Licensee-initiated changes to the ODCM:

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

1. Sufficient information to support the changes together with the appropriate analyses or evaluations justifying the changes and

2. A determination that the changes 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 do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b) Shall become effective after approval of the plant manager.

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 to 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, and shall indicate the date (month and year) the change was implemented.

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

6.9.1.1 Not Used

6.9.1.2 Occupational Radiation Exposure Report

NOTE

A single submittal may be made for the South Texas Project. The submittal should combine sections that are common to both units.

This report provides a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was required, receiving an annual deep dose equivalent greater than 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). The tabulation may include individuals for whom monitoring was provided but not required. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

6.9.1.3 Annual Radiological Environmental Operating Report

NOTE

A single submittal may be made for the South Texas Project. The submittal should combine sections that are common to both units.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 15 of each year. The report shall include summaries, interpretations, and analysis 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 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

6.9.1.4 <u>Radioactive Effluent Release Report</u>

NOTE

A single submittal may be made for the South Texas Project, which shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted by May 1 of each year. 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 be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.9.1.5 <u>Monthly Operating Reports</u>

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.

6.9.1.6 <u>Core Operating Limits Report (COLR)</u>

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

1. Safety limits for thermal power, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) for Specification 2.1,

2. Limiting Safety System Settings for Reactor Coolant Flow-Low Loop design flow, Overtemperature ΔT, and Overpower ΔT setpoint parameter values for Specification 2.2,

3. SHUTDOWN MARGIN limits for Specification 3/4.1.1.1,

4. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3/4.1.1.3,

5. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

6. Control Bank Insertion Limits for Specification 3/4.1.3.6,

7. Axial Flux Difference limits and target band for Specification 3/4.2.1,

(continued)

6.9.1.6.a (continued)

8. Heat Flux Hot Channel Factor, K(Z), Power Factor Multiplier, and (F_{xy}^{RTP}) for Specification 3/4.2.2,

9. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3, and

10. DNB related parameters for Reactor Coolant System T_{avg} Pressurizer Pressure, and the Minimum Measured Reactor Coolant System Flow for Specification 3/4.2.5.

The COLR shall be maintained available in the Control Room.

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:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.1 - Shutdown Margin, Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 - DNB Parameters.)

2. WCAP-12942-P-A, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the South Texas Project Electric Generating Station Units 1 and 2."

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

3. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary Class 2)

(Methodology for Specification 2.1 - Safety Limits, and 2.2 - Limiting Safety System Settings)

4. WCAP-8385, "Power Distribution and Load Following Procedures Topical Report," September 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control.)

(continued)

6.9.1.6.b (continued)

5. Westinghouse letter NS-TMA-2198, T.M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422, January 1981, Docket Nos. 50-369 and 50-370.)

6. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

7. WCAP-10266-P-A, Rev. 2, WCAP-11524-NP-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J.N., et al., March 1987; including Addendum 1-A, "Power Shape Sensitivity Studies," December 1987 and Addendum 2-A, "BASH Methodology Improvements and Reliability Enhancement," May 1988.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

8. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

9. CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

(Methodology for operating at a RATED THERMAL POWER of 3,853 Mwt)

10. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (W Proprietary)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

(continued)

6.9.1.6 (continued)

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

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

6.9.2 Not Used

6.10 Not Used

6.11 Not Used

6.12.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 20.1601(a), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is greater than 100 mrem/h but equal to or less than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., radiation protection technician) 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 radiation levels equal to or less than 1000 mrem/h, 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:

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

b. A radiation monitoring device which 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 determined and entry personnel are knowledgeable of them.

c. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area.

d. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   i. Be under the surveillance of 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 in the RWP, or

   ii Be under surveillance by means of closed circuit television or equivalent, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

(continued)

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to individuals with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) but less than 500 Rads in one hour at one meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked or continuously guarded doors to prevent unauthorized entry. The keys to the doors shall be maintained under the administrative control of the shift manager on duty or radiation protection manager. Doors shall remain locked except during periods of access by individuals under an approved RWP. Prior to entry, individuals shall be informed of the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by individuals qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas, accessible to personnel, with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) but less than 500 Rads in one hour at one meter 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, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 Not Used

6.14 Not Used