Exhibit B

Prairie Island Nuclear Generating Plant

License Amendment Request Dated March 17, 1986
Proposed Changes to the
Technical Specifications Appendix A of
Operating License DPR-42 and DPR-60

Exhibit B consists of revised pages of Appendix A Technical Specifications as listed below:

Table of Contents; i thru viii

Section 1.0; TS.1-1 thru TS.1-8

Section 2.0; TS.2.1-1, TS.2.2-1, TS.2.3-1 thru TS.2.3-4

Section 3.0; TS.3.0-1, TS.3.1-1 thru TS.3.1-13, TS.3.2-1 thru TS.3.2-3, TS.3.3-1 thru TS.3.3-7, TS.3.4-1 thru TS.3.4-3, TS.3.5-1, TS.3.6-1 thru TS.3.6-4, TS.3.7-1 thru TS.3.7-3, TS.3.8-1 thru TS.3.8-4, TS.3.9-1 thru TS.3.9-8, TS.3.10-1 thru TS.3.10-8, TS.3.11-1, TS.3.12-1, TS.3.13-1 thru TS.3.13-2, TS.3.14-1 thru TS.3.14-4, TS.3.15-1

Section 4.0; TS.4.1-1, TS.4.4-3, TS.4.5-4, TS.4.6-1 thru TS.4.6-3, TS.4.7-1, TS.4.8-2, TS.4.9-1, TS.4.11-2, TS.4.13-3, TS.4.14-1, TS.4.15-1, TS.4.16-2, TS.4.17-4, TS.4.18-1

Section 6.0; TS.6.7-1 thru TS.6.7-4

BASES

Section 2.0; Cover page, B.2.1-1, B.2.1-2, B.2.2-1, B.2.3-1, thru B.2.3-3

Section 3.0; Cover Page, B.3.0-1, B.3.1-1, thru B.3.1-12, B.3.2-1, B.3.3-1 thru B.3.3-4, B.3.4-1, B.3.5-1 thru B.3.5-4, B.3.6-1 thru B.3.6-2, B.3.7-1 thru B.3.7-2, B.3.8-1 thru B.3.8-3, B.3.9-1 thru B.3.9-5, B.3.10-1 thru B.3.10-10, B.3.11-1, B.3.12-1, B.3.13-1, B.3.14-1 thru B.3.14-2, B.3.15-1

Section 4.0; Cover Page, B.4.1-1 thru B.4.1-2, B.4.2-1, B.4.3-1, B.4.4-1 thru B.4.4-4, B.4.5-1 thru B.4.5-2, B.4.6-1 thru B.4.6-2, B.4.7-1, B.4.8-1, B.4.9-1, B.4.10-1, B.4.11-1, B.4.12-1 thru B.4.12-2, B.4.13-1, B.4.14-1, B.4.15-1, B.4.16-1 thru B.4.16-2, B.4.17-1 thru B.17-2, B.4.18-1

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TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

1.0 DEFINITIONS

Frequently used terms are defined so that a uniform interpretation of the specifications may be achieved.

Auxiliary Building Special Ventilation Zone Integrity

Auxiliary Building Special Ventilation Zone Integrity exists when:

1. Single doors in the Auxiliary Building Special Ventilation Zone are locked closed, and
2. At least one door in each Auxiliary Building Special Ventilation Zone air lock type passage is closed, and
3. The valves and actuation circuits that isolate the Auxiliary Building Normal Ventilation System following an accident are operable.

Channel Check

Channel check is a qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

A channel functional test consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is operable, including alarm and/or trip initiating action.

Channel Calibration

Channel calibration consists of the adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall include the channel functional test.

Channel Response Test

A channel response test consists of injecting a simulated signal into the channel as near the sensor as practicable to measure the time for electronics and relay actions, including the output scram relay.

Containment Integrity

Containment Integrity exists when:

1. Non-automatic containment isolation valves are either locked closed or they are under direct administrative control and are capable of being closed within one minute following an accident.

2. Blind flanges required by Table TS.4.4-1 are installed.

3. The equipment hatch is closed and sealed.

4. Automatic containment isolation valves are operable.

5. At least one door in each personnel air lock is closed.

Cold Shutdown

A reactor is in the cold shutdown condition when the reactor is subcritical by at least 1% k/k and the reactor coolant average temperature is less than 200°F.

Core Alteration

Core alteration is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel, which may affect core reactivity. Suspension of core alteration shall not preclude completion of movement of a component to a safe conservative position.

Degree Of Instrumentation Redundancy

Degree of instrumentation redundancy is defined as the difference between the number of operable channels and the minimum number of channels which when tripped will cause an automatic shutdown.

Dose Equivalent I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (uCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-Average Disintegration Energy

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

Fire Suppression Water System

The fire suppression water system consists of: Water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

Gaseous Radwaste Treatment System

The Gaseous Radwaste Treatment System is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

Hot Shutdown

A reactor is in the hot shutdown condition when the reactor is subcritical by an amount greater than or equal to the margin as specified in Figure TS.3.10-1 and the reactor coolant average temperature is 547°F or greater.

Limiting Safety System Settings

Limiting safety system settings are settings on protective instrumentation that initiate automatic protective action at a level such that safety limits will not be exceeded.

Limiting Condition(s) For Operation

Limiting conditions for operation are those restrictions on unit operation listed in Section 3 that must be met in order to assure safe operation of the unit.

Members of the General Public

Means all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational occupational, or other purposes not associated with the plant.

Minimum Pressurization Temperature (MPT)

Reactor coolant system temperature below which reactor coolant system pressure is limited by Figures TS.3.1-1 and TS.3.1-2, Reactor Coolant System Heatup and Cooldown Limitations.

Offsite Dose Calculation Manual (ODCM)

The ODCM is the manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents, in the calculation of liquid and gaseous effluent monitoring instrumentation alarm and/or trip setpoints, and in the conduct of environmental radiological environmental monitoring.

Operable - Operability

A system, subsystem, train, component or device shall be Operable or have Operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, 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).

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this paragraph.

The operability of a system or component shall be considered to be established when: (1) it satisfies the Limiting Conditions for Operation in Specification 3.0, (2) it has been tested periodically in accordance with Specification 4.0 and has met its performance requirements, and (3) its condition is consistent with the two paragraphs above.

Physics Tests

Physics tests are those tests that measure fundamental characteristics of the core and related instrumentation. Physics tests are conducted such that the core power is sufficiently reduced to allow for the perturbation due to the test and therefore avoid exceeding power distribution limits in Specification 3.10.B.

Low power physics tests are run at reactor powers less than 5% of rated power.

Power Operation

Power operation of a unit is any operating condition that results when the reactor of that unit is critical, and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Process Control Program (PCP)

The PCP is the manual containing the current formula, sampling, analysis, test and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10CFR20, 10CFR71, and Federal and State regulations and other requirements governing the disposal of radioactive wastes.

Protection Instrumentation and Logic

1. Protection System

   The protection system consists of both the reactor trip system and the engineered safety feature system. The protection system encompasses all electrical and mechanical devices and circuitry (from sensors through the actuating devices) which are required to operate in order to produce the required protective function. Tests of protection systems will be considered acceptable when overlapped if run in parts.

2. Protection System Channel

   A protection system channel is an arrangement of components and modules as required to generate a single protective action signal when required by a unit condition. The channel loses its identity where single action signals are combined.

3. Logic Channel

   A logic channel is a group of relay contact matrices which operate in response to analog channel signals to generate a protective action signal.

Purging

Purging is the 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.

Quadrant Power Tilt

Quadrant power tilt is the ratio of the maximum quadrant power indicated by an upper excore detector to the average reactor power indicated by the upper excore detectors or the ratio of the maximum quadrant power indicated by a lower excore detector to the average reactor power indicated by the lower excore detectors, whichever is greater. Power is proportional to excore detector current times its calibration factor. Percentage quadrant power tilt is 100 times the amount the quadrant power tilt ratio exceeds one.

Rated Thermal Power

Rated thermal power of a unit is the total reactor core heat transfer rate to the reactor coolant of 1650 megawatts thermal (MWt) from the reactor core of that unit.

Refueling

The plant is in refueling when there is fuel in the reactor vessel and the vessel head closure bolts are less than fully tensioned or the head is removed. When in refueling, K_{eff} shall be less than or equal to 0.95 and the reactor coolant average temperature shall be less than or equal to 140°F.

Reportable Event

A Reportable Event shall be any plant occurrence or event which must be reported, per 10 CFR 50.73, requiring written reports to the Commission.

Safety Limits

Safety limits are the quantitative restrictions placed upon reactor operations in order to assure the integrity of the fuel cladding and primary system. If any safety limit is exceeded, the associated unit shall be shut down until the NRC authorizes resumption of operation.

Shield Building Integrity

Shield Building Integrity exists when:

1. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed, and

2. The shield building equipment opening is closed.

Site Boundary

The Site Boundary is a border within which the land is owned, leased, or otherwise controlled by the licensee. The site boundary for liquid releases of radioactive material is defined in Figure 3.9-1. The site boundary for gaseous releases of radioactive material is defined in Figure 3.9-2.

Solidification

Solidification is the conversion of wet radioactive wastes into a form that meets shipping and burial ground requirements.

Source Check

A source check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

Startup Operation

The process of heating up a reactor above 200°F, making it critical, and bringing it up to power operation.

Thermal Power

Thermal power is the total reactor core heat transfer rate to the reactor coolant.

Unrestricted Areas

Means any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

Ventilation Exhaust Treatment System

A Ventilation Exhaust Treatment System is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered safety feature atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment System Components.

Venting

Venting is the 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 not provided or required during venting.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, reactor coolant system pressure and coolant temperature during operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure TS.2.1-1. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system.

Specification

The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, and pressurizer level.

Objective

To provide for automatic protective action in the event that the principal process variables approach a SAFETY LIMIT.

Specification

A. Protective instrumentation settings for reactor trip shall be as follows:

1. Startup protection

   a. High flux, intermediate range (high set point) - current equivalent to $\leq$40% of RATED THERMAL POWER.

   b. High flux, power range (low set point) - $\leq$25% of RATED THERMAL POWER.

   c. High flux, source range - neutron flux $\leq 10^6$ counts/second.

2. Core protection

   a. High flux, power range (high set point) - $\leq$108% of RATED THERMAL POWER.

   b. High pressurizer pressure - $\leq$2385 psig.

   c. Low pressurizer pressure - $\geq$1815 psig.

   d. Overtemperature ΔT

   $$\Delta T_t \leq \Delta T_o \left[K_1 - K_2 (T-T') \left(\frac{1+t_1 s}{1+t_2 s}\right) + K_3(P-P') - f(\Delta I)\right]$$

   where

   ΔT_o = Indicated ΔT at RATED THERMAL POWER
   T = Average temperature, °F
   T' = 567.3°F
   P = Pressurizer pressure, psig
   P' = psig 2235
   K_1 $\leq$ 1.11
   K_2 = 0.0090
   K_3 = 0.000566
   t_1 = 30 sec
   t_2 = 4 sec

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Specification continued

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chamber, with gains to be selected based on measured instrument response during plant startup tests, such that were qt and qb are the percent power in the top and bottom halves of the core, respectively, and qt + qb is total core power in percent of rated power:

1. for $q_t - q_b$ within -12% and -9%, f (ΔI) = 0, and

2. for each percent that the magnitude of $q_t - q_b$ exceeds +9% the ΔT trip set point shall be automatically reduced by an equivalent of 2.5 percent of RATED THERMAL POWER.

3. for each percent that the magnitude of $q_t - q_b$ exceeds -12%, the ΔT trip set point shall be automatically reduced by an equivalent of 1.5 percent of RATED THERMAL POWER.

e. Overpower ΔT

$$\Delta T_p \leq \Delta T_o \left[K_4 - \frac{K_5 \tau_3 sT}{1 + \tau_3 s} - K_6(T-T') - f\ (\Delta I) \right]$$

where

ΔT_o = Indicated ΔT at RATED THERMAL POWER
T = Average temperature, °F
T' = 567.3°F
K_4 $\leq$ 1.10
K_5 = 0.0275 for increasing T; 0 for decreasing T
K_6 = 0.002 for $T > T'$, 0 for $T < T'$
τ_3 = 10, sec
$f(\Delta I)$ = as defined in d. above

f. Low reactor coolant flow per loop - $\geq$90% of normal indicated loop flow as measured at loop elbow tap.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Specification continued

g. Open reactor coolant pump motor breaker.

1. Reactor coolant pump bus undervoltage - $\geq$75% of normal voltage.

2. Reactor coolant pump bus underfrequency - $\geq$58.2HZ

h. Power range neutron flux rate.

1. Positive rate - $\leq$15% of RATED THERMAL POWER with a time constant $\geq$2 seconds

2. Negative rate - $\leq$7% of RATED THERMAL POWER with a time constant $\geq$2 seconds

3. Other reactor trips

a. High pressurizer water level - $\leq$90% of narrow range instrument span.

b. Low-low steam generator water level - $\geq$5% of narrow range instrument span.

c. Low steam generator water level - $\geq$15% of narrow range instrument in coincidence with steam/feedwater mismatch flow - $\leq 1.0 \times 10^6$ lbs/hr.

d. Turbine Generator trip

1. Turbine stop valve indicators - closed

2. Low auto stop oil pressure - $\geq$45 psig

e. Safety injection - See Specification 3.5

B. Protective instrumentation settings for reactor trip interlocks shall be as follows:

1. "At power" reactor trips that are blocked at low power (low pressurizer pressure, high pressurizer level, and loss of flow for one or two loops) shall be unblocked whenever:

a. Power range neutron flux is $\geq$12% of RATED THERMAL POWER or,

b. Turbine load is $\geq$10% of full load turbine impulse pressure.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Specification continued

2. Low power block of single loop loss of flow is permitted whenever power range neutron flux is ≤10% of RATED THERMAL POWER.

3. Power range high flux low setpoint trip and intermediate range high flux trip shall be unblocked whenever power range neutron flux is ≤9% of RATED THERMAL POWER.

4. Source range high flux trip shall be unblocked whenever intermediate range neutron flux is $\leq 10^{-10}$ amperes.

5. Reactor trip on turbine trip shall be unblocked whenever power range neutron flux is ≥50% of RATED THERMAL POWER.

C. Control Rod Withdrawal Stops

1. Block automatic rod withdrawal:

   a. Turbine load ≤15% of full load turbine impulse pressure.

3. LIMITING CONDITIONS FOR OPERATION

3.0 Applicability

A. Compliance with the LIMITING CONDITIONS FOR OPERATION contained in the following Technical Specifications is required during the conditions specified. Upon failure to meet the LIMITING CONDITIONS FOR OPERATION, the associated action requirements shall be met.

B. Noncompliance with a specification shall exist when the requirements of the LIMITING CONDITION FOR OPERATION and associated action requirements are not met within the specified time interval. If the LIMITING CONDITION FOR OPERATION is satisfied prior to expiration of the specified time interval, completion of the action requirements is not required.

C. When a LIMITING CONDITION FOR OPERATION is not met, and required action is not specified or cannot be satisfied, within one hour initiate the action necessary to place the affected unit in Hot Shutdown and be in:

1. At least HOT SHUTDOWN within the next 6 hours, and

2. COLD SHUTDOWN within the following 30 hours.

Where corrective measures are completed such that operation under actions specified in the LIMITING CONDITION FOR OPERATION is once again possible, those actions may be taken in lieu of actions specified above. Time limitations specified by the applicable LIMITING CONDITION FOR OPERATION actions are measured from the time of discovery of the failure to meet the LIMITING CONDITION FOR OPERATION.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system when irradiated fuel is in the containment.

Objective

To specify those LIMITING CONDITIONS FOR OPERATION of the reactor coolant system which must be met to assure safe reactor operation.

Specification

A. Operational Components

1. Reactor Coolant Loops and Coolant Circulation

a. Reactor Critical

A reactor shall not be made or maintained critical unless both reactor coolant pumps are in operation, except during low power PHYSICS TESTS.

b. Reactor Coolant System Average Temperature Above 350°F.

(1) Reactor coolant system average temperature shall not exceed 350°F unless both reactor coolant pumps are OPERABLE with at least one in operation*. If these conditions cannot be satisfied, except as specified in 3.1.A.1.b(2) below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the next 12 hours.

(2) A reactor coolant pump may be inoperable for 72 hours provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, place the affected unit in HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the next 12 hours.

*Both pumps may be shutdown for up to one hour provided the reactor is sub-critical, the reactor trip breakers are open, no operations are permitted that would cause dilution of the reactor coolant boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

c. Reactor Coolant System Average Temperature Below 350°F (and Reactor Coolant Level Above the Reactor Vessel Flange)

(1) Whenever the reactor coolant system average temperature is below 350°F, except during REFUELING SHUTDOWN with the vessel head unbolted, at least two methods for removing decay heat shall be OPERABLE with one in operation*, except as specified in 3.1.A.1.c.(2) below. Acceptable methods for removing decay heat are at least one reactor coolant pump and its associated steam generator; or a residual heat removal loop including a pump and its associated heat exchanger.

(2) With only one OPERABLE method of removing decay heat, initiate action to restore two OPERABLE methods of removing decay heat. If the remaining operable method is an RHR loop, be in COLD SHUTDOWN within 24 hours.

(3) With no OPERABLE methods of removing decay heat, suspend all operations involving a reduction in boron concentration of the reactor coolant system and initiate action to restore one OPERABLE method of removing decay heat.

(4) A reactor coolant pump may be started at RCS temperature less than the MINIMUM PRESSURIZATION TEMPERATURE only if either of the following conditions is met -

There is a steam or gas bubble in the pressurizer, or

The (steam generator minus RCS) temperature difference for the steam generator in that loop is less than 50°F.

d. Reactor Coolant Level Below or at the Reactor Vessel Flange

(1) Both residual heat removal loops, each consisting of a pump and its associated heat exchanger, shall be OPERABLE with one in operation*, except as specified in 3.1.A.3.d.(2) below.

(2) With one or both residual heat removal loop(s) inoperable, action shall be taken to restore the inoperable residual heat removal loop(s) to an OPERABLE status.

*All pumps may be shutdown for up to one hour provided the reactor is subcritical, no operations are permitted that would cause dilution of the reactor coolant boron concentration and core outlet temperature is maintained at least 10°F below saturation temperature.

2. Steam Generator

   a. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless both steam generators are operable. If these conditions cannot be satisfied, except as specified in 3.1.A.2.b below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

   b. During STARTUP OPERATION or POWER OPERATION, one steam generator may be inoperable for 72 hours provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

   c. In the event of excessive primary to secondary leakage take the actions required by Specification 3.1.C.2.e.

3. Reactor Coolant System Pressure Control

a. Pressurizer

(1) A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless there is a steam bubble in the pressurizer and heater groups A and B are operable. If these conditions cannot be satisfied, except as specified in 3.1.A.3.a.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor system average temperature below 350°F within the following 12 hours.

(2) During STARTUP OPERATION or POWER OPERATION, Group A or B pressurizer heater group may be inoperable for 72 hours provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

b. Pressurizer Safety Valves

(1) Reactor Coolant System average temperature greater than or equal to 350°F.

A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless two pressurizer safety valves are OPERABLE, with lift settings of 2485 psig ±1%. If these conditions cannot be satisfied, discontinue STARTUP OPERATION and within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

(2) Reactor Coolant System Average Temperature below 350°F

At least one pressurizer safety valve shall be OPERABLE, with a lift setting of 2485 psig ±1%, whenever the head is on the reactor vessel, except during hydrostatic tests. With no pressurizer safety valve OPERABLE, place an OPERABLE residual heat removal loop into operation.

c. Pressurizer Power Operated Relief Valves

(1) A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless two power operated relief valves (PORVs) and their associated block valves are OPERABLE. If these conditions cannot be satisfied, except as specified in 3.1.A.3.C.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

(2) During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored or the block valve cannot be closed within the time specified, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

(a) With one or more PORVs inoperable, within one hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s).

(b) With one or more block valves inoperable, within one hour either restore the block valve(s) to OPERABLE status or close the valve.

(3) With Reactor Coolant System temperature less than Minimum Pressurization Temperature, both pressurizer power operated relief valves (PORVs) shall be OPERABLE with the over pressure protection system enabled, the associated block valve open, and the associated backup air supply charged. One PORV may be inoperable for 7 days. If these conditions cannot be met, the reactor coolant system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within 8 hours.

4. Reactor Coolant Vent System

a. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless reactor coolant vent system paths from both the reactor vessel head and pressurizer steam space are OPERABLE and closed except as specified in 3.1.A.4.b and 3.1.A.4.c below.

b. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit until OPERABILITY is restored. If any one of these conditions is not restored to an OPERABLE status within 30 days, the reactor shall be placed in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

(1) Both of the parallel vent valves in the reactor vessel head vent path inoperable, or

(2) Both of the parallel vent valves in the pressurizer vent path inoperable, or

(3) The vent valve to the pressurizer relief tank discharge line inoperable, or

(4) The vent valve to the containment atmospheric discharge line inoperable.

c. With no reactor coolant vent system path OPERABLE, restore at least one vent path to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

B. Pressure/Temperature Limits

1. The Unit 1 and Unit 2 reactor coolant system (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures TS.3.1-1 and TS.3.1-2 during heatup, cooldown, and criticality with:

   a. A maximum heatup of 100°F in any 1-hour period.

   b. A maximum cooldown of 100°F in any 1-hour period.

2. The pressurizer temperature shall be limited to:

   a. A maximum heatup of 100°F in any 1-hour period.

   b. A maximum cooldown of 200°F in any 1-hour period.

3. The pressurizer spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

4. If any of the limits of 3.1.B.1 or 3.1.B.2 are exceeded:

   a. Restore temperature and/or pressure to within acceptable limits within 30 minutes and,

   b. Within one hour, initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce the reactor coolant system average temperature and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

   c. The affected unit shall not be returned to service until an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of reactor coolant system is performed, and it is determined that the Reactor Coolant System remains acceptable for continued operation.

C. REACTOR COOLANT SYSTEM LEAKAGE

1. Leakage Detection

At least two means of reactor coolant system leakage detection shall be OPERABLE, one of which must depend on the detection of radionuclides in the containment. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Leakage Limitations

a. If the leakage rate, from other than controlled leakage sources, such as the reactor coolant pump controlled leakage seals, exceeds 1 gpm and the source of the leakage is not identified within 4 hours of leak detection, place the affected unit at least in HOT SHUTDOWN within the next 6 hours. If the source of leakage is not identified within an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours.

b. If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in 3.1.C.2.c below.

c. If it is determined that leakage exists through a fault which has developed in a reactor coolant system component body, pipe wall, vessel wall, or pipe weld, and that the fault cannot be isolated, within one hour initiate action to place the affected unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and take corrective action prior to resumption of unit operation.

d. If the total leakage, other than leakage from controlled sources, exceeds 10 gpm, within one hour and initiate action to place the affected unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours. If the condition is not corrected within an additional 8 hours, be in COLD SHUTDOWN within the following 30 hours and remain in COLD SHUTDOWN until the condi-ion is corrected.

e. If steam generator leakage between the primary coolant system and the secondary coolant system of a unit exceeds 1.0 gallon per minute (gpm), within one hour initiate action to place the affected unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and perform an inservice steam generator tube inspection in accordance with Technical Specification 4.12.

3. Pressure Isolation Valve Leakage

Leakage through the pressure isolation valves shall not exceed the maximum allowable leakage specified in Specification 4.3 when reactor coolant system average temperature exceeds 200°F. If the maximum allowable leakage is exceeded, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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.

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.3.1-5, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 500°F within the following 6 hours.

   (b) With the specific activity of the primary coolant greater than $100/\bar{E}$ microcurie per gram, within one hour initiate action to place the affected unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 500°F within the following 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 $100/\bar{E}$ 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.C.

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, of 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-' our period, or the transient limits are reached, place the affected unit 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 re-started 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 primary 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.

F. MINIMUM CONDITIONS FOR CRITICALITY

1. The reactor shall be made critical only at or above the coolant temperature at which the following reactivity coefficient is negative and remains negative for any coolant temperature increase (except during low power PHYSICS TESTS):

   (a) Moderator temperature coefficient for a reactor loaded with Westinghouse fuel only.

   (b) Isothermal temperature coefficient for a reactor either full or partially loaded with Exxon fuel.

2. When the reactor coolant temperature is below the minimum temperature as specified in 1. above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to reactor coolant depressurization.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the chemical and volume control system.

Objective

To define those conditions of the chemical and volume control system necessary to assure safe reactor operation and safe COLD SHUTDOWN.

Specification

A. When fuel is in a reactor and reactor coolant system average temperature is at or below 200°F there shall be at least one flow path to the core for boric acid injection. If no OPERABLE flow path exists, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

B. One unit shall not be made or maintained critical nor shall the reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.2.D or 3.2.E below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

1. A minimum of two charging pumps for the unit shall be OPERABLE.

2. Two of the four boric acid transfer pumps shall be OPERABLE if required to meet the requirements of Specification 3.2.B.4.

3. One boric acid tank shall contain a minimum of 2000 gallons of 11.5% to 13% by weight boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be OPERABLE to the extent of establishing two independent flow paths for boric acid injection -- one flow path from the boric acid tanks to the core and one flow path from the refueling water storage tank to the core.

5. Two channels of heat tracing shall be OPERABLE for the flow paths from the boric acid tanks required to meet the requirements of Specification 3.2.B.4.

6. Automatic valves, piping, and interlocks associated with the above components which are required to operate for the steam line break accident are OPERABLE.

7. Motor-operated valve Number 8809C for that unit shall be open, shall have its valve position monitor light OPERABLE, and shall have its motor control center supply breaker physically locked in the open position.

8. Manual valves in the boric acid system shall be physically locked in the position required for automatic boric acid injection following a steam line break accident.

C. Except as specified in 3.2.D, the reactor in the second unit shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F with the reactor in the other unit already critical unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.2.D or 3.2.E below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

1. A minimum of two charging pumps for each unit shall be OPERABLE.

2. At least three of the four boric acid transfer pumps shall be OPERABLE if required to meet the requirements of Specification 3.2.C.4.

3. At least two boric acid tanks shall each contain a minimum of 2000 gallons of 11.5% to 13% by weight boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be OPERABLE to the extent of establishing for each unit two independent flow paths for boric acid injection -- one flow path from its associated boric acid tank to the core and one flow path from its refueling water storage tank to its core. The flow paths shall be arranged so that each boric acid tank can supply only its associated unit.

5. Two channels of heat tracing shall be OPERABLE for the flow paths from the boric acid tanks required to meet the requirements of Specification 3.2.C.4.

6. Automatic valves, piping, and interlocks associated with the above components which are required to operate for the steam line break accident are OPERABLE.

7. The motor-operated valve in each unit numbered 8809C shall be open, shall have its valve position monitor light OPERABLE, and shall have its motor control center supply breaker physically locked in the open position.

8. Manual valves in the boric acid system shall be physically locked in the position required for automatic boric acid injection following a steam line break accident in either unit.

D. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist during the time intervals specified, provided STARTUP OPERATION is discontinued until OPERABILITY is restored except as specified in 3.2.E below. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next six hours. If OPERABILITY is not restored within an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours.

1. One charging pump may be inoperable for 24 hours.

2. When operating both units, one of the three designated boric acid transfer pumps may be inoperable for 24 hours.

3. One channel of heat tracing may be inoperable for 48 hours.

4. Any one redundant automatic valve required for boric acid injection following a steam line break may be inoperable for 24 hours.

5. One of the 2 independent flow paths in each unit for boric acid addition to the core may be inoperable for 7 days. Prior to initiating repairs, the other flow path shall be tested to demonstrate OPERABILITY.

6. A unit may operate for 8 hours with no OPERABLE boric acid storage tank.

E. During plant shutdown, if the boron concentration of the primary system is equivalent to or greater than the COLD SHUTDOWN boron concentration, the requirements of 3.2.B.3 and C.3 are not required to be satisfied.

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.3.A.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 1950 ppm.

   b. Each reactor coolant system accumulator shall be OPERABLE when reactor coolant system pressure is greater than 1000 psig.

      OPERABILITY requires:
      (1) The isolation valve is open
      (2) Volume is between 1250 and 1282.9 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of at least 700 psig

   c. Two safety injection pumps are OPERABLE except that pump control switches in the control room shall meet the requirements of Section 3.3.A.3 whenever the reactor coolant system temperature is less than MINIMUM PRESSURIZATION TEMPERATURE.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers are OPERABLE.

f. Automatic valves, interlocks and piping associated with the above components and required to function during accident conditions, are OPERABLE.

g. Manual valves in the above systems that could (if one is improperly positioned) reduce injection flow below that assumed for accident analyses, shall be blocked and tagged in the proper position for injection. RHR system valves, however, may be positioned as necessary to regulate plant heatup or cooldown rates when the reactor is subcritical. All changes in valve position shall be under direct administrative control.

h. For Unit 1 operation, the following valve conditions shall exist:

(1) Safety injection system motor-operated valves 8801A, 8801B, 8806A shall have valve position monitor lights OPERABLE and shall be locked in the open position by having the motor control center supply breakers physically locked open.

(2) Safety injection system motor-operated valves 8816A and 8816B shall be closed, shall have valve position monitor lights OPERABLE, and shall have the motor control center supply breakers physically locked open.

(3) Accumulator discharge valves 8800A and 8800B shall have position monitor lights and alarms OPERABLE.

(4) Residual Heat Removal System valves 8701A and 8701B shall have normal valve position indication OPERABLE.

i. For Unit 2 operation, the valve conditions corresponding to those stated in Specification 3.3.A.1.h for Unit 1 shall exist.

2. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours. If OPERABILITY is not restored within an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours.

a. One safety injection pump may be inoperable for 72 hours.

b. One residual heat removal pump may be inoperable for 72 hours.

c. One residual heat exchanger may be inoperable for 72 hours.

d. Any redundant valve in the system required for safety injection, may be inoperable for 72 hours.

e. One accumulator may be inoperable for one hour whenever pressurizer pressure is greater than 1000 psig.

f. One safety injection system and one residual heat system may be inoperable for 72 hours provided the redundant safety injection system and heat removal system required for functioning during accident conditions is OPERABLE.

3. At least one safety injection pump control switch in the control room shall be in pullout whenever RCS temperature is less than MINIMUM PRESSURIZATION TEMPERATURE except that both SI pumps may be run for conduct of the integrated SI test when either of the following conditions is met:

(a) There is a steam or gas bubble in the pressurizer and the SI pump discharge valves are shut, or

(b) The reactor vessel head is removed.

B. Containment Cooling Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.3.B.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

a. Two containment spray pumps are OPERABLE.

b. Four containment fan cooler units are OPERABLE.

c. The spray additive tank is OPERABLE with not less than 2590 gallons of solution with a sodium hydroxide concentration of 9% to 11% by weight inclusive.

d. Manual valves in the above systems that could (if improperly positioned) reduce spray flow below that assumed for accident analysis, shall be blocked and tagged in the proper position. During POWER OPERATION, changes in valve position will be under direct administrative control.

e. Automatic valves, interlocks, ducts, dampers, controls and piping associated with the above components and required for accident conditions are OPERABLE.

f. The following motor-operated valve conditions shall exist:

(1) The Unit 1 operation, containment spray system motor-operated valves MV32096 and MV32097 shall be closed and shall have the motor control center supply breakers open.

(2) For Unit 2 operation, containment spray system motor-operated valves MV32108 and MV32109 shall be closed and shall have the motor control center supply breakers open.

2. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours. If OPERABILITY is not restored within an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours.

a. One containment fan cooler unit may be inoperable for 7 days, provided both containment spray pumps are OPERABLE.

b. One containment spray pump may be inoperable for 72 hours, provided at least two containment fan cooler units are OPERABLE.

c. Two containment fan cooler units may be inoperable for 72 hours, provided one containment spray pump is OPERABLE.

d. Two containment spray pumps may be inoperable for 72 hours, provided four containment fan cooler units are OPERABLE.

e. The spray additive tank may be inoperable for 72 hours.

C. Component Cooling Water System

1. Single Unit Operation

a. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F, unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.3.C.1.b below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(1) The two component cooling pumps assigned to that unit are OPERABLE.

(2) The two component cooling heat exchangers assigned to that unit are OPERABLE.

(3) All valves, interlocks, instrumentation and piping associated with the above components, and required for the functioning of the system during accident conditions, are OPERABLE.

b. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist provided startup operation is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours. If OPERABILITY is not restored within an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours.

(1) One of the assigned component cooling pumps may be inoperable for 72 hours.

(2) One of the assigned component cooling heat exchangers may be inoperable for 72 hours.

2. Two-Unit Operation

a. A second reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F, unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.3.C.2.b below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(1) Three component cooling pumps are OPERABLE.

(2) Four component cooling heat exchangers are OPERABLE.

(3) All valves, interlocks and piping associated with the above components, and required for the functioning of the system during accident conditions, are OPERABLE.

b. During STARTUP OPERATIONS or POWER OPERATION either one of the following conditions of inoperability may exist provided STARTUP OPERATION is discontinued until operability is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours. If OPERABILITY is not restored within an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours.

(1) One of the three component cooling pumps may be inoperable for 72 hours.

(2) One of the two component cooling heat exchangers associated with each unit may be inoperable for 72 hours.

D. Cooling Water System

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F, unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.3.D.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   a. Two diesel-driven cooling water pumps and two motor-driven cooling water pumps are OPERABLE.

   b. All valves, interlocks, instrumentation, piping and fuel oil supply required for the functioning of the cooling water system during accident conditions are OPERABLE.

2. During STARTUP OPERATION or POWER OPERATION, the following conditions of inoperability may exist provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   a. One diesel-driven cooling water pump may be inoperable for 7 days (total for both diesel-driven cooling water pumps during any consecutive 30 day period) provided:

      (1) the other diesel-driven pump and its associated diesel generator are OPERABLE.

      (2) the engineered safety features associated with the operable diesel-driven cooling water pump are OPERABLE; and

      (3) both paths from transmission grid to the plant 4 kV safeguards buses are OPERABLE.

      (4) two motor-driven cooling water pumps shall be OPERABLE.

   b. One of the two required motor-driven cooling water pumps may be inoperable for 7 days provided both diesel-driven cooling water pumps are OPERABLE.

   c. One of the two required cooling water headers may be inoperable for 72 hours provided:

      (1) the diesel-driven pump and the diesel generator associated with safety features on the operable header are OPERABLE.

      (2) the horizontal motor-driven pump associated with the OPERABLE header and the vertical motor-driven pump are OPERABLE.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the steam and power conversion system.

Objective

To specify minimum conditions of steam-relieving capacity and auxiliary feed-water supply necessary to assure that capability of removing decay heat from the reactor, and to limit the concentration of activity that might be released by steam relief to the atmosphere.

Specification

A. Safety and Relief Valves

A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the following conditions are satisfied. If these conditions cannot be satisfied within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor system average temperature below 350°F within the following 12 hours.

1. Rated relief capacity of ten steam system safety valves is available for that reactor, except during testing.

2. Both steam generator power-operated relief valves for that reactor are OPERABLE.

B. Auxiliary Feed System

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the following conditions are satisfied. If these conditions cannot be satisfied, except as specified in 3.4.B.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

    a. For single unit operation, the turbine-driven pump associated with that reactor plus one motor-driven pump are OPERABLE.

    b. For two-unit operation, all four auxiliary feedwater pumps are OPERABLE.

c. Valves and piping associated with the above components are OPERABLE except that during STARTUP OPERATION necessary changes may be made in motor-operated valve position. All such changes shall be under direct administrative control.

d. A minimum of 100,000 gallons of water is available in the condensate storage tanks and a backup supply of river water is available through the cooling water system.

e. For Unit 1 operation motor operated valves MV32242 and MV32243 shall have valve position monitor lights OPERABLE and shall be locked in the open position by having the motor control center supply breakers manually locked open. For Unit 2, corresponding valve conditions shall exist.

f. Manual valves in the above systems that could (if one is improperly positioned) reduce flow below that assumed for accident analysis shall be locked in the proper position for emergency use. During POWER OPERATION, changes in valve position will be under direct administrative control.

g. The condensate supply cross connect valves C-41-1 and C-41-2, to the auxiliary feedwater pumps shall be blocked and tagged open. Any changes in position of these valves shall be under direct administrative control.

2. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within an additional 48 hours, place the affected unit (or either unit in the case of a motor driven AFW pump inoperability) in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 12 hours.

   a. A turbine driven AFW pump may be inoperable for 72 hours.

   b. A motor driven AFW pump may be inoperable for 72 hours.

C. Steam Exclusion System

1. The reactor coolant system average temperature shall not exceed 350°F unless both isolation dampers in each ventilation duct penetrating rooms containing equipment required for a high energy line rupture outside of containment are operable, except as specified below.

   a. If one of the two redundant steam exclusion dampers is inoperable, the operable redundant damper may remain open for 24 hours. If after 24 hours, the damper remains inoperable, one of the two dampers shall be closed.

   b. The actuation logic for one train of steam exclusion may be inoperable for 24 hours. If after 24 hours, the actuation logic remains inoperable, one of the two dampers shall be closed.

2. If two redundant steam exclusion dampers or two trains of actuation logic are inoperable, close the associated dampers within 4 hours.

D. Radiochemistry

A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the specific activity of the secondary coolant system for that reactor is less than or equal to 0.10 uCi/gm DOSE EQUIVALENT I-131. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor system coolant average temperature below 350°F within the following 12 hours.

3.5 INSTRUMENTATION SYSTEM

Applicability

Applies to protection system instrumentation.

Objectives

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded, and to delineate the conditions of the reactor trip and engineered safety feature instrumentation necessary to ensure reactor safety.

Specification

A. Limiting set points for instrumentation which initiates operation of the engineered safety features shall be as stated in Table TS.3.5-1.

B. For on-line testing or in the event of failure of a sub-system instrumentation channel, plant operation shall be permitted to continue at RATED THERMAL POWER in accordance with Tables TS.3.5-2 through TS.3.5-6.

C. If the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum OPERABLE Channels, or if the specified Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in the column titled Operator Action of Tables TS.3.5-2 through TS.3.5-6.

D. In the event of sub-system instrumentation channel failure permitted by Specification 3.5.B, the requirements of Tables TS.3.5-2 through TS.3.5-6 need not be observed during the short period of time the OPERABLE sub-system channels are tested where the failed channel must be blocked to prevent unnecessary reactor trip. If the test time exceeds four hours, operation shall be limited according to the requirement shown in the column titled Operator Action of Tables TS.3.5-2 through TS.3.5-6.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of the containment system.

Objective

To define the operating status of the containment system for plant operation.

Specification

A. Containment Integrity

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless CONTAINMENT INTEGRITY is maintained.

2. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. Vacuum Breaker System

1. Both valves in each of two vacuum breaker systems, including actuating and power circuits, shall be OPERABLE when CONTAINMENT INTEGRITY is required except as specified in 3.6.B.2 below. If these conditions cannot be satisfied, except as specified in 3.6.B.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. One vacuum breaker system may be inoperable for 7 days.

C. Containment Isolation Valves

1. Automatic containment isolation valves, listed in Table TS.4.4-1 including actuation circuits, shall be OPERABLE when CONTAINMENT INTEGRITY is required except as specified in 3.6.C.2 below. If these conditions cannot be satisfied, except as specified in 3.6.C.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more valve(s) listed in Table TS.4.4-1 inoperable, within four hours:

   (a) restore the inoperable valve(s) to operable status or,

   (b) deactivate the operable valve in the closed position or,

   (c) lock closed at least one valve in each penetration having one inoperable valve.

D. Containment Purge System

1. The 36-inch containment purge system double gasketed blind flanges shall be installed whenever the reactor is above COLD SHUTDOWN. The 18-inch containment inservice purge system double gasketed blind flanges shall be installed whenever the reactor is above COLD SHUTDOWN except as noted below. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. The inservice purge system may be operated above COLD SHUTDOWN if the following conditions are met:

   a. The debris screens are installed on the supply and exhaust ducts in containment.

   b. Both valves shall satisfactorily pass a local leak rate test prior to use.

   c. The two automatic primary containment isolation valves and the automatic shield building ventilation damper in each duct that penetrates containment shall be OPERABLE, including instruments and controls associated with them.

   d. The blind flanges (i.e., 42B (53 in Unit 2) and 43A (52 in Unit 2) shall be reinstalled and satisfactorily pass a local leak rate test, each time after the inservice purge system is used.

E. Auxiliary Building Special Ventilation Zone Integrity

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY is maintained. If these conditions cannot be satisfied, except as specified in 3.6.E.2 and 3 below, within one hour initiate the action necessary to place both units in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Openings in the Auxiliary Special Ventilation Zone are permitted provided they are under direct administrative control and they can be reduced to less than 10 square feet within 6 minutes following an accident.

3. Valves and actuation circuits that isolate the Auxiliary Building Normal Ventilation System following an accident may be inoperable for 7 days provided the ventilation system can be manually isolated within 6 minutes following an accident.

F. Auxiliary Building Special Ventilation System

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless both trains of the Auxiliary Building Special Ventilation System are OPERABLE. In order for the Auxiliary Building Special Ventilation System to be considered OPERABLE, the Turbine Building roof exhauster fans shall be capable of being deenergized within 30 minutes following a loss-of-coolant accident. If these conditions cannot be satisfied, except as specified in 3.F.2 below, within one hour initiate the action necessary to place both units in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. One train of the Auxiliary Building Special Ventilation System may be inoperable for 7 days.

G. Shield Building Integrity

A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless SHIELD BUILDING INTEGRITY is maintained. If these conditions cannot be satisfied, within 24 hours initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

H. Shield Building Ventilation System

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless both trains of the Shield Building Ventilation System are

OPERABLE. If these conditions cannot be satisfied, except as specified in 3.H.2 and 3.H.3 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. One train of the Shield Building Ventilation System may be inoperable for 7 days.

3. Both trains of the Shield Building Ventilation System may be inoperable for 24 hours.

I. Containment Internal Pressure

1. The internal pressure of the containment vessel shall not exceed 2 psig whenever CONTAINMENT INTEGRITY is required except as specified in 3.6.I.2 below.

2. If internal pressure exceeds 2 psig and is not corrected within 8 hours, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

J. Containment and Shield Building Air Temperature

1. The average temperature of the air in the containment vessel shall not exceed 44°F above the average temperature of the air in the shield building whenever CONTAINMENT INTEGRITY is required except as specified in 3.6.J.2 below.

2. If this limit is exceeded and is not corrected within 8 hours, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

K. Containment Shell Temperature

1. Containment Shell Temperature shall be equal to or greater than 30°F whenever CONTAINMENT INTEGRITY is required except as specified in 3.6.K.2 below.

2. If this limit is exceeded and is not corrected within 8 hours, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

L. Electric Hydrogen Recombiners

1. Both containment hydrogen recombiner systems shall be OPERABLE whenever the reactor is above HOT SHUTDOWN. If these conditions cannot be satisfied, except as specified in 3.6.L.2 below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours.

2. One hydrogen recombiner system may be inoperable for 30 days.

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

Applies to the availability of electrical power for the operation of plant auxiliaries.

Objectives

To define those conditions of electrical power availability necessary to assure safe reactor operation and continuing availability of engineered safeguards.

Specification

A. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless all of the following requirements are satisfied for the applicable unit. If these conditions cannot be satisfied, except as specified in 3.7.B below, within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

1. At least two separate paths from the transmission grid to the plant 4 kV safeguards buses, each capable of providing adequate power to minimum safety related equipment, shall be OPERABLE.

2. The 4 kV safeguards buses of that unit, 15 and 16 or 25 and 26, shall be energized.

3. The 480 V safeguards buses for that unit, 110 and 120 or 210 and 220, and their safeguards motor control centers shall be energized.

4. Reactor protection instrument AC buses for that unit shall be energized: 111, 112, 113, and 114 or 211, 212, 213, and 214.

5. D1 and D2 diesel generators are OPERABLE, and a fuel supply of 70,000 gallons is available in the interconnected storage tanks for the diesel generators and the diesel driven cooling water pump diesel engines.

6. Both batteries with their associated chargers and both d-c safeguard systems shall be OPERABLE.

7. No more than one of the Instrument AC Panels 111, 112, 113, and 114 shall be powered from Panel 117 or its associated instrument inverter bypass source.

8. No more than one of the Instrument AC Panels 211, 212, 213, and 214 shall be powered from Panel 217 or its associated instrument inverter bypass source.

B. During STARTUP OPERATION or POWER OPERATION, any of the following conditions of inoperability may exist for the times specified, provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

1. One path from the grid to the plant 4 kV safeguards buses may be inoperable for 7 days provided (a) D1 and D2 diesel generators are OPERABLE, and (b) all engineered safety features equipment is OPERABLE.

2. Two paths from the grid to the plant 4 kV safeguards buses may be inoperable for 12 hours provided, (a) the D1 and D2 diesel generators are already operarating or are demonstrated to be OPERABLE sequentially performing Surveillance requirement 4.6.A.1.e on each diesel within 8 hours, and (b) all engineered safety features equipment is OPERABLE.

3. One path from the grid to the plant 4 kV safeguards buses and one diesel generator may be inoperable for 12 hours provided, (a) the OPERABILITY of the other diesel generator is demonstrated* by performance of Surveillance Requirement 4.6.A.1.e within 8 hours** and (b) all engineered safety features equipment associated with the OPERABLE diesel generator is OPERABLE.

4. One diesel generator may be inoperable for 7 days (total for both diesel generators during any consecutive 30 day period) provided (a) the OPERABILITY of the other diesel generator is demonstrated* by performance of surveillance requirement 4.6.A.2.e within 24 hours**, (b) all engineered safety features equipment associated with the operable diesel generator is OPERABLE and (c) both paths from the grid to the plant 4 kV safeguards buses are OPERABLE.

5. D1 and D2 diesel generators may be inoperable for 2 hours provided both paths from the grid to the plant 4 kV safeguards buses are OPERABLE.

* The OPERABILITY of the other diesel generator need not be demonstrated if the diesel inoperability was due to preplanned preventative maintenance or testing.

** This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

6. One 4 kV safeguards bus (and its associated 480 V bus including associated safeguards motor control centers) or one 480 V safeguards bus including associated safeguards motor control centers may be inoperable or not fully energized for 8 hours provided its redundant counterpart is demonstrated OPERABLE and the diesel generator and safeguards equipment associated with its counterpart are OPERABLE.

7. One battery charger may be inoperable provided, (a) its associated battery is OPERABLE, (b) its redundant counterpart is demonstrated OPERABLE, and (c) the diesel generator and safeguards equipment associated with its counterpart are OPERABLE.

8. One battery may be inoperable for 8 hours provided that the other battery and both battery chargers remain OPERABLE.

9. In addition to the requirements of Specification TS.3.7.A.7 a second inverter supplying Instrument AC Panels 111, 112, 113, and 114 may be powered from an inverter bypass source for 8 hours.

10. In addition to the requirements of Specification TS.3.7.A.8 a second inverter supplying Instrument AC Panels 211, 212, 213, and 214 may be powered from an inverter bypass source for 8 hours.

3.8 REFUELING AND FUEL HANDLING

Applicability

Applies to operating limitations associated with fuel-handling operations, CORE ALTERATIONS, and crane operations in the spent fuel pool enclosure.

Objectives

To ensure that no incident could occur during fuel handling, CORE ALTERATIONS and crane operations that would affect public health and safety.

Specification

A. Core Alterations

1. During CORE ALTERATIONS the following conditions shall be satisfied except as specified in 3.8.A.2 below:

   a. The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, at least one isolation valve or equivalent shall be operable or locked closed in each line which penetrates the containment and provides a direct path from containment atmosphere to the outside.

   b. Radiation levels in the fuel handling areas of the containment shall be monitored continuously.

   c. The core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment, which are in service whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

   d. During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration of the reactor coolant system and the refueling cavity shall be sufficient to ensure that the more restrictive of the following reactivity conditions is met: $K_{eff} \leq 0.95$ or the boron concentration ≥ 2000 ppm. The required boron concentration shall be verified by chemical analysis daily.

e. During movement of fuel assemblies or control rods out of the reactor vessel, at least 23 feet of water shall be maintained above the reactor vessel flange. The required water level shall be verified prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.

f. At least one residual heat removal pump shall be OPERABLE and running. The pump may be shut down for up to one hour to facilitate movement of fuel or core components.

g. If the water level above the top of the reactor vessel flange is less than 20 feet, except for control rod unlatching/ latching operations or upper internals removal/replacement, both residual heat removal loops shall be OPERABLE.

h. Direct communication between the control room and the operating floor of the containment shall be available whenever CORE ALTERATIONS are taking place.

i. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 100 hours.

j. The radiation monitors which initiate isolation of the Containment Purge System shall be tested and verified to be OPERABLE prior to a CORE ALTERATIONS.

2. If any of the above conditions cannot be met, take the appropriate action listed below:

   a. If any of the above conditions are not met, CORE ALTERATIONS shall cease. Work shall be initiated to correct the violated conditions so that the specifications are met, and no operations which may increase the reactivity of the core shall be performed.

   b. If Specification 3.8.A.f or 3.8.A.g cannot be satisfied, all fuel handling operations in containment shall be suspended, the requirements of Specification 3.8.A.1.a shall be satisfied, and no reduction in reactor coolant boron concentration shall be made.

B. Fuel Handling and Crane Operation

1. During fuel handling operations or crane operation with loads over the spent fuel pools (inside the spent fuel pool enclosure), the following conditions shall be satisfied:

   a. Radiation levels in the spent fuel storage pool area shall be monitored continuously during fuel handling operations.

   b. Prior to introducing a spent fuel shipping cask into the spent fuel pool area:

      (1) A minimum boron concentration of 1800 ppm shall be maintained in spent fuel pools No. 1 and 2. The required boron concentration shall be verified by chemical analysis daily while use of the cask continues, and

      (2) A cask impact limiter determined to be capable of absorbing the impact energy of a cask drop or a crash pad capable of absorbing the impact energy of a cask drop shall in in place, and

      (3) Crane interlocks and mechanical stops limiting travel to the approved load path shall be determined to be OPERABLE, and

      (4) Fuel in the small pool (pool No. 1) shall have been discharged from a reactor for at least 5 years.

   c. Prior to spent fuel handling in the auxiliary building, tests shall be made to determine the OPERABILITY of the spent fuel pool special ventilation system including the radiation monitors in the normal ventilation system that actuate the special system and isolate the normal systems.

   d. Prior to fuel handling operations, fuel-handling cranes shall be load-tested for OPERABILITY of limit switches, interlocks, and alarms.

   e. When the spent fuel cask contains one or more fuel assemblies, it will not be suspended more than 30 feet above any surface until the fuel has decayed more than 90 days.

   f. Both trains of the Spent Fuel Pool Special Ventilation System shall be OPERABLE. A train of the Spent Fuel Pool Special Ventilation System shall be considered operable only if its associated diesel generator is operable.

2. If any of the conditions in 3.8.B.1, above, cannot be met, suspend all fuel handling operations and crane operations with loads over the spent fuel pools (inside the spent fuel pool enclosure) except as specified in 3.8.B.2.a below:

   a. One train of Spent Fuel Pool Ventilation may be inoperable for 7 days.

C. No more than 45 recently discharged assemblies shall be located in the small pool (pool No. 1).

3.9 RADIOACTIVE EFFLUENTS

Applicability

Applies at all times to the liquid and gaseous radioactive effluents from the plant and the SOLIDIFICATION and packaging for shipment of solid radioactive waste.

Objective

To implement the requirements of 10CFR20, 10CFR71, 10CFR50 Section 50.36a, Appendix A and Appendix I to 10CFR50, 40CFR141, and 40CFR190 pertaining to radioactive effluents.

Specifications

A. Liquid Effluents

1. Concentration

   a. The concentration of liquid radioactive material released at any time from the site (Figure 3.9-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} uci/ml total activity.

   b. When the concentration of radioactive material in liquid released from the site exceeds the limits in (a) above, immediately restore the concentration within acceptable limits.

2. Dose

   a. The dose or dose commitment to an individual from radioactive materials in liquid effluents released from the site (Figure 3.9-1) shall be limited:

      1. During any calendar quarter to ≤ 3.0 mrem to the total body and to ≤ 10 mrem to any organ, and

      2. During any calendar year to ≤ 6 mrem to the total body and to ≤ 20 mrem to any organ.

b. When the calculated dose from the release of radioactive materials in liquid released from the site to UNRESTRICTED AREAS exceeds the limits in (a) above, in lieu of any other report prepare and submit to the Commission a special report within 30 days which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

3. Liquid Radwaste System

a. The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected dose due to liquid effluent released from the site (Figure 3.9-1) when averaged over one month would exceed 0.12 mrem to the total body or 0.4 mrem to any organ.

b. With radioactive liquid waste being discharged without treatment and in excess of the limits in (a) above, within 30 days submit to the Commission a special report which includes the following information:

1. Identification of the inoperable equipment or sub-systems and the reason for inoperability.

2. Action(s) to be taken to restore equipment to OPERABLE status, and

3. Summary description of action(s) taken to prevent a recurrence.

4. Liquid Storage Tanks

a. The quantity of radioactive material 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

b. With the quantity of radioactive material in any of the above listed tanks exceeding the limit in (a) above, immediately suspend all additions of radioactive materials to the tank and within 48 hours reduce the tank contents to within the limit.

B. Gaseous Effluents

1. Dose Rate

a. The dose rate at any time due to radioactive materials released in gaseous effluents from the site (Figure 3.9-2) shall be limited to the following values:

1. The dose rate limit for noble gases shall be $\leq$500 mrem/year to the total body and $\leq$3000 mrem/year to the skin, and

2. The dose rate limit for I-131, tritium, and radioactive particulates with half-lives greater than eight days shall be $\leq$1500 mrem/year to any organ.

b. With the dose rate(s) exceeding the limits in (a) above, immediately decrease the release rate to within acceptable limits.

2. Dose from Noble Gases

a. The air dose in UNRESTRICTED AREAS due to noble gases released in gaseous effluents from the site (Figure 3.9-2) shall be limited to the following values:

1. During any calendar quarter, to $\leq$10 mrad for gamma radiation and $\leq$20 mrad for beta radiation, and

2. During any calendar year, to $\leq$20 mrad for gamma radiation and $\leq$40 mrad for beta radiation.

b. With the calculated air dose from radioactive noble gases in gaseous affluent exceeding any of the above limits, within 30 days submit to the Commission a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective action(s) taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

3. Dose from I-131, Tritium, and Radioactive Particulate With Half-Lives Greater Than Eight days.

   a. The dose to any organ of an individual due to I-131, tritium, and radioactive particulates with half-lives greater than eight days released in gaseous effluents from the site (Figure 3.9-2) shall be limited to the following:

      1. During any calendar quarter to ≤15 mrem, and

      2. During any calendar year to ≤30 mrem

   b. With the calculated dose from the release of I-131, tritium. and radioactive particulates with half-lives greater than eight days in gaseous effluents exceeding the limit(s) in (a) above, in lieu of any other report prepare and submit to the Commission a special report within 30 days which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

4. Gaseous Radwaste Treatment System and Ventilation Exhaust Treatment Systems

   a. The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEMS shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected dose due to gaseous effluents released from the site (Figures 3.9-2) when averaged over one month would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose.

   b. With gaseous waste being discharged without full treatment and in excess of the limits in (a) above, within 30 days submit to the Commission a special report which includes the following information:

1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

2. Action(s) taken to restore the inoperable equipment to operable status, and

3. Summary description of action(s) taken to prevent a recurrence.

c. Except as provided for in (d) below, the concentration of oxygen at the outlet of each operating recombiner shall be limited to $\leq$2% by volume.

d. With the concentration of oxygen measured at the outlet of operating recombiner(s) >2% by volume but $\leq$4% by volume, restore the concentration of oxygen to $\leq$2% by volume within 48 hours.

e. With the concentration of oxygen at the outlet of operating recombiner(s) >4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to $\leq$2% within one hour.

f. The quantity of radioactivity contained in each gas storage tank shall be limited to $\leq$78,800 curies of noble gases (considered as dose equivalent Xe-133).

g. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

h. The radioactive gas contained in the waste gas holdup system shall not be deliberately discharged to the environment during unfavorable wind conditions when the cooling towers are in operation. For the purposes of this specification, unfavorable wind conditions are defined as wind from 5 west of north to 45° east of north at 10 miles per hour or less.

5. Containment Purging

a. Containment PURGE and VENT releases shall be treated during power operation through the charcoal and particulate filters of the in-service purge system or shield building ventilation system. |

b. Prior to PURGING containment during POWER OPERATION or immediately after shutdown if the containment is to be purged, the sampling and analysis specified in Table 4.17-4 shall be completed. |

C. Solid Radioactive Waste

1. A solid radwaste system shall be operable and used, as applicable in accordance with a PROCESS CONTROL PROGRAM for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 Prior to shipment of radioactive wastes from the site.

2. With the packaging requirements of 10 CFR Part 20 or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

D. Dose from All Uranium Fuel Cycle Sources

1. The dose or dose commitment to a MEMBER OF THE GENERAL PUBLIC from all uranium fuel cycle sources is limited to ≤25 mrem to the total body or any organ (except for the thyroid, which is limited to ≤75 mrem) over a period of 12 consecutive months.

2. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.9.A.2.a.1, 3.9.A.2.a.2, 3.9.B.2.a.1, 3.9.B.2.a.2, 3.9.B.3.a.1, or 3.9.B.3.a.2, submit within 30 days a special report to the Commission which calculates the highest radiation exposure to any MEMBER OF THE GENERAL PUBLIC from all uranium fuel cycle sources (including all affluent pathways and direct radiation). Unless this report shows that exposures are less than the 40 CFR Part 190 standard, either apply to the Commission for a variance to continue releases which exceed the 40 CFR Part 190 standard or reduce subsequent releases to permit the standard to be met.

E. Radioactive Liquid Effluent Monitoring Instrumentation

1. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.9-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.A.1.a are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

2. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specifications, immediately suspend the release of radioactive liquid effluents monitored by the effected channel or declare the channel inoperable.

3. With less than the minimum required radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the action shown in Table 3.9-1.

F. Radioactive Gaseous Effluent Monitoring Instrumentation

1. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.9-2 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifi cation 3.9.B.1.a are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

2. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

3. With less than the minimum required radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the action shown in Table 3.9-2.

G. The provisions of specification 3.0.C are not applicable.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distribution and to the limits on control rod operations.

Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

A. Shutdown Reactivity

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for PHYSICS TESTS, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN conditions if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no chenges in xenon or boron concentration.

B. Power Distribution Limits

1. At all times, except during low power PHYSICS TESTING, measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, as defined below and in the bases, shall meet the following limits:

$$F_Q^N \times 1.03 \times 1.05 \leq (2.32/P)^* \times K(Z) \times BU(E_j)$$

$$F_{\Delta H}^N \times 1.04 \leq 1.55 \times [1 + 0.2(1-P)]$$

where the following definitions apply:

(a) K(Z) is the axial dependence function shown in Figure TS.3.10-5.

(b) Z is the core height location.

(c) E_j is the maximum pellet exposure in fuel rod j for which the F_Q^N is being measured.

(d) $BU(E_j)$ is the normalized exposure dependence function for Exxon Nuclear Company fuel shown in Figure TS.3.10-7. For Westinghouse fuel, $BU(E_j) = 1.0$

(e) P is the fraction of RATED THERMAL POWER at which the core is operating. In the F_Q^N limit determination when $P \leq .50$, set $P = 0.50$.

*(2.21/P) shall be used for Westinghouse assemblies.

(f) F_Q^N or $F_{\Delta H}^N$ is defined as the measured F_Q^N or $F_{\Delta H}^N$ respectively, with the smallest margin or greatest excess of limit.

(g) 1.03 is the engineering hot channel factor, F_Q^E, applied to the measured F_Q^N to account for manufacturing tolerance.

(h) 1.05 is applied to the measured F_Q^N to account for measurement uncertainty.

(i) 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty.

2. Hot channel factors, F_Q^N and $F_{\Delta H}^N$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

(a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

(b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux differance was last determined, by 10% or more of RATED THERMAL POWER.

F_Q^N (equil) shall meet the following limit for the middle axial 80% of the core:

$$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (2.32/P) \times K(Z) \times BU(E_j)$$

where V(Z) is defined Figure 3.10-8 and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N or $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

(b) If the measured F_Q^N (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N (equil) x 1.03 x 1.05 x V(Z) exceeds the (2.32/P) x K(Z) x BU(E_j) limit.

(c) If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a HOT SHUTDOWN condition with return to power authorized up to 50% of RATED THERMAL POWER for the purpose of PHYSICS TESTING. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above 50% of RATED THERMAL POWER. THERMAL POWER may then be increased provided F_Q^N or $F_{\Delta H}^N$ is demonstrated through in-core mapping to be within its limits.

(d) If two successive measurements indicate an increase in the peak pin power $F_{\Delta H}^N$ with exposure, either of the following actions shall be taken:

1. F_Q^N (equil) shall be multiplied by 1.02 x V(Z) x 1.03 x 1.05 for comparison to the limit specified in 3.10.B.2, or

2. F_Q^N (equil) shall be measured at least once per seven effective full power days until two successive maps indicate that the peak pin power, $F_{\Delta H}^N$, is not increasing.

4. Except during PHYSICS TESTS, and except as provided by specifications 5 through 8 below, the indicated axial flux difference for at least three operable excore channels shall be maintained within a ±5% band about the target flux difference.

5. Above 90 percent of RATED THERMAL POWER:

If the indicated axial flux difference of two OPERABLE excore channels deviates from its larger band, within 15 minutes either eliminate such deviation, or reduce THERMAL POWER to less than 90 percent of RATED THERMAL POWER.

6. Between 50 and 90 percent of RATED THERMAL POWER:

a. The indicated axial flux difference may deviate from its ±5% target band for a maximum of one* hour (cumulative) in any 24-hour period provided that the difference between the indicated axial flux difference about the target flux difference does not exceed the envelope shown in Figure TS.3.10-6.

b. If 6.a is violated for two OPERABLE excore channels then the THERMAL POWER shall be reduced to less than 50% of RATED THERMAL POWER and the high neutron flux setpoint reduced to less than 55% of RATED THERMAL POWER.

*May be extended to 16 hours during incore/excore calibration.

c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of at least three OPERABLE excore channels being within the target band.

7. Less than 50 percent of RATED THERMAL POWER:

   a. The indicated axial flux difference may deviate from its target band.

   b. A power increase to a level greater than 50 percent of RATED THERMAL POWER is contingent upon the indicated axial flux difference of at least three OPERABLE excore channels not being outside the target band for more than one hour (cumulative) out of the preceding 24 hour period.

8. In applying 6a and 7b above, penalty deviations outside the ±5% target band shall be accumulated on a time basis of:

   a. One minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

   b. One-half minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

9. If alarms associated with monitoring the indicated axial flux difference deviations from the ±5% target band are not operable, the indicated axial flux difference value for each OPERABLE excore channel shall be logged at least once per hour for the first 24 hours and half-hourly thereafter until the alarms are returned to an OPERABLE status. For the purpose of applying this specification, logged values of indicated axial flux difference must be assumed to apply during the previous interval between loggings.

C. Quadrant Power Tilt Limits

1. Except for physics tests, if the percentage QUADRANT POWER TILT exceeds 2% but is less than 7%, the rod position indication shall be monitored and logged once each shift to verify rod position within each bank assignment and, within two hours, one of the following steps shall be taken:

   a. Correct the tilt to less than 2%

   b. Restrict core power level so as not to exceed RATED POWER, less 2% for every percent that the QUADRANT POWER TILT ratio exceeds 1.0.

2. If the percentage QUADRANT POWER TILT exceeds 2% but is less than 7% for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for PHYSICS TESTS if the QUADRANT POWER TILT ratio exceeds 1.07, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel inoperable, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion; insertion limits are shown in Figure TS.3.10-2, -3 and -4 for normal and abnormal operating conditions.

3. Control bank insertion may be further restricted by specification 3.10.A if, (1) the measured control rod worth of all rods, less the worth of the worst stuck rod, is less than 5.52% reactivity at the beginning of the first cycle or the eqivalent value if measured at any other time, or (2) if a rod is inoperable (Specification 3.10.G).

4. Insertion limits do not apply during PHYSICS TESTS or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for low power PHYSICS TESTING. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power PHYSICS TEST.

E. Rod Misalignment Limitations

1. If a full-length rod cluster control assembly (RCCA) is misaligned from its bank by more than 24 steps, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.

2. a. If the bank demand position is greater than or equal to 215 steps, or less than or equal to 30 steps and the rod position indicator channel differs by more than 24 steps, that rod control cluster assembly (RCCA) shall be considered misaligned.

   b. If the bank demand position is between 30 and 215 steps and the rod position indicator channel differs by more than 12 steps, that RCCA shall be considered misaligned.

3. If the misaligned RCCA is not realigned within a total of 8 hours, the RCCA shall be declared inoperable.

F. Inoperable Rod Position Indicator Channels

1. If a rod position indicator (RPI) channel is out of service then

   a. For operation between 50% and 100% of RATED THERMAL POWER, the position of the RCCA shall be checked directly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to rod motion exceeding a total of 24 steps, whichever occurs first.

   b. During operation below 50% of RATED THERMAL POWER, no special monitoring is required.

2. The plant shall be brought to the HOT SHUTDOWN Condition should more than one RPI channel per group or more than two RPI channels per bank be found to be inoperable during POWER OPERATION.

3. If a full length rod having a rod position indicator channel inoperable is found to be misaligned from 1.a. above, then apply Specification 3.10.E.

G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under specification 3.10.E. or 3.10.H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

2. The Plant shall be brought to the hot shutdown condition should more than one inoperable full length rod be discovered during power operation.

3. If the inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in Figure TS.3.10-3 apply.

4. If the inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in Figure TS.3.10-4 apply.

5. If POWER OPERATION is continued with one inoperable full-length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is earlier made OPERABLE. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, THERMAL POWER shall be reduced to a level consistent with the safety analysis.

H. Rod Drop Time

At operating temperature and full flow, the drop time of each full-length RCCA shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.

I. Monitor Inoperability Requirements

1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of RATED THERMAL POWER, and after 30 inches or more of rod motion.

2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of RATED THERMAL POWER in addition to the increased surveillance requirements.

3. If one or both of the QUADRANT POWER TILT monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of RATED THERMAL POWER

J. DNB Parameters

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq 564°F$
b. Pressurizer Pressure ≥ 2220 psia*
c. Reactor Coolant Flow $\geq 178,000$ gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours. Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

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*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

3.11 CORE SURVEILLANCE INSTRUMENTATION

Applicability

Applies to the OPERABILITY of the moveable detector instrumentation system and the core thermocouple instrumentation system.

Objective

To specify OPERABILITY requirements for the moveable detector and core thermocouple systems.

Specification

A. The moveable detector system shall be OPERABLE following the initial fuel loading and each subsequent reloading so that the power distribution can be confirmed. If the system is degraded to the extent that a full core map cannot be obtained, the measurement error allowance due to incomplete mapping shall be substantiated by the licensee.

B. A minimum of 2 moveable detector thimbles per quadrant, and sufficient detectors, drives, and readout equipment to map these thimbles, shall be operable during recalibration of the excore axial offset detection system per Specification 4.1. If this operability for recalibration of excore nuclear instruments when required by Specification 4.1 cannot be achieved, power shall be limited to 90% of RATED THERMAL POWER until recalibration is completed in accordance with this specification.

C. A minimum of 4 thermocouples or 2 moveable detectors per quadrant shall be operable for readout if the reactor is operated above 85% of RATED THERMAL POWER with one excore nuclear power channel inoperable (see Specification 3.10 C.4).

D. The provisions of specification 3.0.C are not applicable.

3.12 SNUBBERS

Applicability

Applies to the OPERABILITY of safety related snubbers.

Objective

To define those conditions of snubber OPERABILITY necessary to assure safe reactor operation.

Specification

A. Except as permitted below, all safety related snubbers shall be OPERABLE above COLD SHUTDOWN. Snubbers may be inoperable in COLD SHUTDOWN and REFUELING whenever the supported system is not required to be OPERABLE.

B. With one or more snubbers made or found to be inoperable for any reason when OPERABILITY is required, within 72 hours:

   1. Replace or restore the inoperable snubbers to OPERABLE status and perform an engineering evaluation per Specification 4.13.E on the supported component(s), or

   2. Declare the supported system inoperable and take the action required by the Technical Specifications for inoperability of that system.

3.13 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the OPERABILITY of the Control Room Special Ventilation System.

Objective

To specify OPERABILITY requirements for the Control Room Special Ventilation System.

Specification

A. Control Room Special Ventilation System

1. Both trains of the Control Room Special Ventilation System shall be OPERABLE at all times. If these conditions cannot be satisfied, except as specified in 3.13.A.2 below, within one hour initiate the action necessary to place both units in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and CORE ALTERATIONS/ fuel handling operations shall be terminated within two hours.

2. With one train of the Control Room Special Ventilation System inoperable, POWER OPERATION or CORE ALTERNATIONS/fuel handling operations are permissible only during the succeeding 7 days. If OPERABILITY is not restored within 7 days, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and CORE ALTERATIONS/fuel handling operations shall be terminated within two hours.

B. Chlorine Detection Systems

Two independent chlorine detection systems, each consisting of two channels of instrumentation shall be OPERABLE at all times except as specified below. The alarm/trip setpoint shall be adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm.

1. If one chlorine detection channel for one train of ventilation is inoperable, then within 7 days:

   a. Restore the inoperable channel to Operable status, or

   b. Operate the redundant ventilation system in the normal (non-recirculation) mode, and close the outside air supply dampers for the affected train of ventilation.

2. If both chlorine detection channels for one train of ventilation are inoperable then within 6 hours:

   a. Restore at least one channel to OPERABLE status, or

   b. Operate the redundant ventilation system in the normal (non-recirculation) mode and close the outside air supply dampers for the affected train of ventilation.

3. If all chlorine monitors for both trains of ventilation are inoperable then within 6 hours close all Control Room ventilation outside air supply dampers.

3.14 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability

Applies to instrumentation and plant systems used for fire detection and protection of the nuclear safety-related structures, systems, and components of the plant.

Objective

To insure that the structures, systems, and components of the plant important to nuclear safety are protected from fire damage.

Specification

A. Fire Detection Instrumentation

1. Except as specified below, the minimum fire detection instrumentation for each fire detection zone shown in Table 3.14-1 shall be OPERABLE whenever equipment in that fire detection zone is required to be OPERABLE. Fire detection instruments located within containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

2. If specification 3.14.A.1 cannot be met:

   a. Within one hour, establish a fire watch patrol to inspect the zone with the inoperable instruments at least once per hour. Fire zones located inside primary containment are exempt from this requirement when CONTAINMENT INTEGRITY is required.

   b. Restore the inoperable instruments to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the malfunction and the plans for restoring the instruments to OPERABLE status.

B. Fire Suppression Water System

1. Except as specified in 3.14.B.2 or 3.14.B.3 below, the system shall be operable at all times with:

   a. The following pumps, including automatic initiation logic, operable and capable of delivering at least 2000 gpm at a discharge pressure of 108 psig.

   1. Diesel-driven fire pump
   2. Motor-driven fire pump
   3. Screen wash pump

b. An OPERABLE flow path capable of taking suction from the river and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant valves and the first valve ahead of each deluge valve, hose station, or sprinkler system required to be OPERABLE.

2. With one or two of the pumps required by Specification 3.14.B.1.a inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide a special report to the Commission within 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in the FIRE SUPPRESSION WATER SYSTEM.

3. With the FIRE SUPPRESSION WATER SYSTEM otherwise inoperable:

   a. Establish a backup FIRE SUPPRESSION WATER SYSTEM within 24 hours, and

   b. Provide a special report to the Commission within 30 days outlining the actions taken and the plans and schedule for restoring the inoperable system to OPERABLE status.

C. <u>Spray and Sprinkler Systems</u>

1. Whenever equipment protected by the following spray and sprinkler systems is required to be OPERABLE, the spray and sprinkler system shall be OPERABLE:

   a. Auxiliary Feed Pump Room WP-10

   b. Diesel Generator Areas PA-1

   c. Unit No. 1 Electrical Penetration Area PA-3

   d. Unit No. 1 Electrical Penetration Area PA-4

   e. Unit No. 2 Electrical Penetration Area PA-6

   f. Unit No. 2 Electrical Penetration Area PA-7

   g. Screenhouse PA-9

2. If Specification 3.14.C.1 cannot be met, a continuous fire watch with backup fire suppression equipment shall be established within one hour. Restore inoperable spray and sprinkler systems to OPERABLE status within 14 days or submit a special report to the Commission within 30 days outlining the cause of inoperabiliity and the plans for restoring the system to OPERABLE status.

D. Carbon Dioxide System

1. Except as specified in 3.14.D.3 below, the CO_2 system protecting the relay and cable spreading room area shall be OPERABLE with a minimum level of 60% in the CO_2 storage tank.

2. During those periods when the relay and cable spreading room area is normally occupied, automatic initiation of the CO_2 system may be bypassed. During those periods when the area is normally unoccupied, the CO_2 system shall be capable of automatic initiation unless there are personnel actually in the area.

3. If specification 3.14.D.1 cannot be met, a continuous fire watch with backup fire suppression equipment shall be stationed in the relay and cable spreading room within one hour. Restore the system to OPERABLE status within 14 days or submit a special report to the Commission within 30 days outlining the cause of inoperability and the plans for restoring the system to OPERABLE status.

F. Fire Hose Stations

1. Whenever equipment protected by hose stations in the following areas is required to be OPERABLE, the hose station(s) protecting that area shall be OPERABLE:

   a. Diesel Generator Rooms
   b. Safety Related Switchgear Rooms
   c. Safety Related Areas of Screenhouse
   d. Auxiliary Building
   e. Control Room
   f. Relay & Cable Spreading Room
   g. Battery Rooms
   h. Auxiliary Feed Pump Room

2. If Specification 3.14.E.1 cannot be met, within one hour hoses supplied from OPERABLE hose stations shall be made available for routing to each area with an inoperable hose station.

   Restore the inoperable hose station(s) to OPERABLE status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the stations to OPERABLE status.

F. Yard Hydrant Hose Houses

1. Whenever equipment in the following buildings is required to be operable, the yard hydrant hose houses in the main yard loop adjacent to each building shall be operable:

   a. Unit No. 1 Reactor Building

   b. Unit No. 2 Reactor Building

   c. Turbine Building

   d. Auxiliary Building

   e. Screen House

2. If Specification 3.14.F.1 cannot be met, within one hour have sufficient additional lengths of 2-1/2 inch diameter hose located in adjacent OPERABLE yard hydrant hose house(s) to provide service to the unprotected area(s).

   Restore the yard hydrant hose house(s) to OPERABLE status within 14 days or submit a a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the houses to OPERABLE status.

G. Penetration Fire Barriers

1. All penetration fire barriers in fire area boundaries protecting equipment required to be OPERABLE shall be OPERABLE.

2. If Specification 3.14.G.1 cannot be met within one hour:

   a) establish a continuous fire watch on at least one side of the affected penetration(s), or

   b) verify the OPERABILITY of the fire detectors on at least one side of the inoperable barrier and establish an hourly fire watch.

   Restore the inoperable penetration fire barriers to OPERABLE status within 7 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the barriers to OPERABLE status.

H. The provisions of specification 3.0.C are not applicable.

3.15 EVENT MONITORING INSTRUMENTATION

Applicability

Applies to plant instrumentation which does not perform a protective function, but which provides information to monitor and assess important parameters during and following an accident.

Objective

To ensure that sufficient information is available to operators to determine the effects of and determine the course of an accident to the extent required to carry out required manual actions.

Specification

A. Process Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-1 shall be OPERABLE.

2. With the number of OPERABLE event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-1, either restore the inoperable channels to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

3. With the number of OPERABLE event monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table TS.3.15-1, either restore the minimum number of channels to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

B. Radiation Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-2 shall be OPERABLE.

2. With the number of OPERABLE event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-2, either restore the inoperable channels to OPERABLE status within 7 days, or prepare and submit a special report to the Commission within 30 days outlining the action taken, the cause of the inoperability, the plans and the schedule for restoring the system to OPERABLE status.

3. With the number of OPERABLE event monitoring instrumentation channels less than the Minimum Channels Operable requirement of Table TS.3.15-2, initiate the preplanned alternate method of monitoring the appropriate parameters and either restore the inoperable channels to OPERABLE status within 7 days, or prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.7.B.2 within the next 30 days outlining the action taken, the cause of the inoperability, the plans and the schedule for restoring the system to OPERABLE status.

4. The provisions of specification 3.0.C are not applicable.

4.0 SURVEILLANCE REQUIREMENTS

Specific time intervals between tests may be adjusted plus or minus 25% to accommodate normal test schedules with the exception that, the intervals between tests scheduled for refueling shutdowns shall not exceed two years.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to SAFETY LIMITS and LIMITING CONDITIONS FOR OPERATION.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification

A. Calibration, testing, and checking of instrumentation channels and testing of logic channels shall be performed as specified in Table TS.4.1-1.

B. Equipment tests shall be conducted as specified in Table TS.4.1-2A.

C. Sampling tests shall be conducted as specified in Table TS.4.1-2B.

D. Whenever the plant condition is such that a system or component is not required to be operable the surveillance testing associated with that system or component may be discontinued. The asterisked items in Tables 4.1-1, 4.1-2A, and 4.1-2B are required at all times, however. Discontinued surveillance tests shall be resumed less than one test interval before establishing plant conditions requiring OPERABILITY of the associated system or component, unless such testing is not practicable (i.e., nuclear power range calibration cannot be done prior to reaching power operation) in which case the testing will be resumed within 48 hours of attaining the plant condition which permits testing to be accomplished.

B. Emergency Charcoal Filter Systems

1. Periodic tests of the shield building ventilation system shall be performed at quarterly intervals to demonstrate OPERABILITY. Each redundant train shall be initiated from the control room and determined to be OPERABLE at the time of its periodic test if it meets drawdown performance computed for the test conditions with 75% of the shield building inleakage specified in Figure TS 4.4-1 after initiation and achieve a pressure -2.0 inches of water gage.

2. Periodic test of the auxiliary building special ventilation system shall be performed at approximately quarterly intervals to demonstrate its OPERABILITY. Each redundant train shall be initiated from the control room and determined to be OPERABLE at the time of periodic test if it isolates the normal ventilation system and produces a measureable negative pressure in the ABSVZ within 6 minutes after initiation.

3. At least once per operating cycle, or once each 18 months, whichever comes first, tests of the filter units in the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System shall be performed as indicated below:

   a. The pressure drop across the combined HEPA filters and the charcoal adsorbers shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).

   b. The inlet heaters and associated controls for each train shall be determined to be OPERABLE.

   c. Verify that each train of each ventilation system automatically starts on a simulated signal of safety injection and high radiation (Auxiliary Building Special Ventilation only).

4. a. The tests listed below shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

   (1) In-place DOP and halogenated hydrocarbons tests at design flows on HEPA filters and charcoal adsorbers banks respectively shall show $\geq$99% DOP removal for particles having a mean diameter of 0.7 microns and $\geq$99% halogenated hydrocarbons removal.

   (2) Laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

h. Following completion of high head safety injection system or RHR system modifications that alter system flow characteristics a flow balance test shall be performed during shutdown to confirm the following injection flow rates are achieved:

1. High Head Safety Injection System:

(a) Flow through all four injection lines plus miniflow shall not exceed 835 gpm with one pump in operation.

(b) The minimum flow through loop A & B cold legs shall be 670 gpm with one pump in operation. The flow rates in each leg shall be within 20 gpm of each other with one pump in operation.

(c) Flow orifices and throttling valves will be used to limit and balance flow through the reactor vessel injection lines to a maximum of the total flow limit in Specification 4.5.B.3.h.1.(a) above, with one pump in operation. During this flow test the flow rates in each leg shall be within 50 gpm of each other.

2. RHR System:

The minimum flow through each RHR Reactor Vessel Injection line shall be at least 1800 gpm.

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 and engine-mounted tank.

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

   c. Verify that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 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 to synchronous speed (900 rpm) with generator voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz. Subsequently, manually sychronize the generator, gradually load* to at least 1650 kW, and operate for at least 60 minutes.

2. At least once each 6 months, for each diesel generator:

   a. Verify the diesel generator starts* and accelerates to at least 900 rpm in less than or equal to 10 seconds.

   b. Verify the generator voltage and frequency to be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal.

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

c. Manually synchronize the generator, load to at least 1650 kW in less than or equal to 60 seconds and operate for at least one hour.

3. At least once each 18 months:

   a. Subject each diesel generator to a thorough inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service.

   b. For each unit, simulate a loss of offsite power in conjunction with a safety injection signal, and:

      1. Verify de-energization of the emergency buses and load shedding from the emergency buses.

      2. Verify the diesels start* on the auto-start signal and energize the emergency buses in one minute.

      3. Verify that the auto-connected loads do not exceed 3000 kw.

   c. Verify the capability of each generator to operate at least one hour while loaded to 3000 kw.

   d. Verify the capability of each generator to reject a load of at least 650 kw without tripping.

   e. During this test, operation of the emergency lighting system shall be ascertained.

   f. Verify that the diesel generator system trips, except those for engine overspeed, ground fault, and generator differential current, are automatically bypassed.

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures.

B. Station Batteries

1. Each battery shall be tested each month. Tests shall include measuring voltage of each cell to the nearest hundredth volt, and measuring the temperature and density of a pilot cell in each battery.

2. The following additional measurements shall be made every three months: the density and height of electrolyte in every cell, the amount of water added to each cell, and the temperature of each fifth cell.

3. All measurements shall be recorded and compared with previous data to detect signs of deterioration or need of equalization charge according to the manufacturer's recommendation.

4. The batteries shall be subjected to a performance test discharge during the first refueling and once every five years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

5. Integrity of Station Battery fuses shall be checked once each day when the battery charger is running.

C. Pressurizer Heater Emergency Power Supply

The emergency pressurizer heater supply shall be demonstrated OPERABLE at least once every 18 months by transferring Backup Heater Group "B" from its normal bus to its safeguards bus and energizing the heaters.

4.7 MAIN STEAM ISOLATION VALVES

Applicability

Applies to periodic testing of the main steam isolation valves.

Objective

To verify the ability of the main steam isolation valves to close upon signal.

Specification

The main steam isolation valves shall be tested during REFUELING. A closure time of five seconds or less shall be verified.

C. Steam Exclusion System

Isolation dampers in each duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be tested for OPERABILITY once each month.

In addition, damper mating surfaces shall be examined visually once each year to assure that no physical change has occurred that could affect leakage.

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, submit a special report to the Commission within 30 days.

D. Tests resulting in 0.005 microcuries or more of removable contamination on the test sample shall be reported to the Commission on an annual basis.

E. Plant operating records shall be made as follows:

1. An inventory of licensed radioactive materials in possession shall be maintained current at all times.

2. The following records shall be retained for 2 years:

   a. Test results in microcuries, for tests performed pursuant to TS 4.11.

   b. Record of annual physical inventory verifying accountability of sources on record.

If any snubber selected for functional testing either fails to lockup or fails to move (i.e., frozen in place) the cause shall be evaluated and all snubbers subject to the same defect shall be functionally tested. This testing is in addition to the regular sample and specified re-samples.

D. Hydraulic snubber functional tests shall verify that:

a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

E. An engineering evaluation shall be performed for all components supported by inoperable snubbers. The purpose of this engineering evaluation shall be to determine if the components were adversely affected by the inoperable snubber(s) to ensure that the components remain capable of meeting the designed service.

F. The installation and maintenance records for each snubber shall be reviewed at least once every 18 months to verify that the indicated service life will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement, or reconditioning shall be indicated in the records.

4.14 CONTROL ROOM AIR TREATMENT SYSTEM TESTS

Applicability

Applies to the periodic testing requirements for the Control Room Special Ventilation System.

Objective

To specify tests for assuring the OPERABILITY of the Control Room Special Ventilation System.

Specification

A. At least once per operating cycle or once every 18 months, whichever occurs first, the following shall be demonstrated:

1. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate (±10%).

2. Automatic initiation of the Control Room Special Ventilation System shall be demonstrated with a simulated high radiation or Safety Injection signal.

B. 1. The following tests shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

a. The results of in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks respectively shall show $\geq$99% DOP removal for particles having a mean diameter of 0.7 microns and $\geq$99% halogenated hydro-carbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

c. Fans shall be shown to operate within ±10% of 4000 cfm.

2. 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 effect the HEPA bank bypass leakage.

3. 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.

4. Each circuit shall be operated at least 15 minutes every month.

4.15 SPENT FUEL POOL SPECIAL VENTILATION SYSTEM

Applicability

Applies to the periodic testing requirements for the Spent Fuel Pool Special Ventilation System (SFPSVS).

Objective

To specify tests for assuring the OPERABILITY of the Spent Fuel Pool Special Ventilation System.

Specification

A. At least once per operating cycle or once every 18 months, whichever occurs first, the following shall be demonstrated:

   1. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate (±10%).

   2. Automatic initiation of each train shall be demonstrated with a simulated high radiation signal.

B. 1. The following tests shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

      a. The results of in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks respectively shall show $\geq$99% DOP removal for particles having a mean diameter of 0.7 microns and $\geq$99% halogenated hydrocarbon removal.

      b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

      c. The Spent Fuel Pool Special Ventilation System fans shall operate within ±10% of 5200 cfm per train.

   2. 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.

   3. 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.

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

b. The motor-driven fire pump shall be started every month and run for at least 15 minutes on recirculation flow.

c. The diesel-driven fire pump shall be started every month from ambient conditions and run for at least 20 minutes on recirculation flow.

d. The level in the diesel-driven fire pump fuel storage tank shall be checked every month and verified to contain at least 500 gallons of fuel.

e. Every three months verify that a sample of fuel from the diesel-driven fire pump fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

f. Every 18 months subject the diesel-driven fire pump engine to an inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service.

g. A simulated automatic actuation of each fire pump and the screen wash pump, including verification of pump capability, shall be conducted every 18 months.

h. The header system shall be flushed every 12 months.

i. System flow tests shall be performed every three years.

j. Valves in flow paths supplying fire suppression water to safety related structures, systems, and components shall be cycled every 12 months.

k. Each valve (manual, power operated, or automatic) in the flow path for safety-related areas and areas posing a fire hazard to safety-related areas, shall be verified to be in its correct position every month and the method of securing the valve in its correct position shall be verified every month.

C. Solid Radioactive Waste

1. Verification of Solidification

a. The PROCESS CONTROL PROGRAM (PCP) shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and chemical solutions).

b. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PCP, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative. SOLIDIFICATION parameters determined by the PCP.

c. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PCP shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PCP shall be modified as required, as provided for in Section 6 of the Technical Specifications.

D. Dose from All Uranium Fuel Cycle Sources

a. Cumulative dose contributions from all plant liquid and gaseous effluents shall be determined in accordance with Specifications 4.17.A.2.a, 4.17.B.2.a, 4.17.B.3.a, and the methods in the ODCM.

4.18 REACTOR COOLANT VENT SYSTEM PATHS

Applicability

Applies to the surveillances performed on the reactor coolant vent system paths to verify OPERABILITY.

Objective

To assure that the capability exists to vent noncondensible gases from the reactor coolant system that could inhibit natural circulation core cooling.

Specification

A. Vent Path Operability

Each reactor coolant vent system path shall be demonstrated OPERABLE prior to commencing STARTUP OPERATION after each refueling by:

1. Verifying all manual isolation valves in each vent path are blocked and tagged in the open position.
2. Cycling each solenoid operated valve in the vent paths through at least one complete cycle of full travel from the control room.

B. System Flow Testing

Flow shall be verified through each reactor coolant vent system path following each refueling.

6.7 Reporting Requirements

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)

This report shall cover the previous calendar year. The report should tabulate on an annual basis 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. The dose assignment to various duty functions may be estimates 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 shall be assigned to specific major work functions.

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.

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.

3. Monthly Operating 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.

4. Semiannual Radioactive Effluent Release Report

Routine radioactive effluent release reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1st and July 1st of each year.

The radioactive effluent release reports 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 to be submitted 60 days after January 1 of each year shall include an assessment of the radiation doses from radioactive effluents released from the plant during the previous calendar year. This same 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 these reports. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM) or standard NRC computer codes.

The report to be submitted 60 days after January 1 of each year 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 release reports 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 releases reports 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.

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.

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:

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 sha include summaries, interpretations, and an analysis of tre is of the results of the radiological environmental surveillance activities for the report period, including a comparison with pre-operational 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

BASES FOR

SECTION 2.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.1 SAFETY LIMIT, REACTOR CORE

Bases

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNB ratio of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions (Reference 1).

The safety limit curves of Figure TS.2.1-1 define the regions of acceptable operation with respect to average temperatures, power, and pressurizer pressure. These boundaries of acceptable operations are limited by the thermal overpower limit (fuel melting), thermal overtemperature limit (cladding damage based on DNB considerations), and the locus of points where the steam generation safety valves open. These limits are used to set the overpower and overtemperature ΔT trip setpoints.

For the overtemperature limit, the following four limiting criteria are used:

1. Vessel exit temperature <650°F (design temperature limit).

2. Vessel exit temperature < saturation temperature (ensures power ~ ΔT).

3. MDNBR >1.3 (fuel damage limit).

4. Hot channel exit quality <15% (limit on CHF correlations).

The first two criteria result in a single limit on vessel exit temperature. For the 1685 psig and 1985 psig curves, the coolant average enthalpy at the core exit is equal to saturated water enthalpy below

2.1 SAFETY LIMIT, REACTOR CORE

Bases continued

power levels of 91% and 74% respectively. For the 2235 psig and 2385 psig curves, the coolant average temperature at the core exit is equal to 650°F below power levels of 64% and 73% respectively.

The third and fourth criteria area evaluated using standard DNB methodology. For all four curves the DNBR is equal to 1.3 at higher power levels. The area of safe operation is below these curves.

The plant conditions required to violate the limits in the lower power range are precluded by the self-actuated safety valves on the steam generators. The highest nominal setting of the steam generator safety valves is 1129 psig (saturation temperature 560°F). At zero power the difference between primary coolant and secondary coolant is zero and at full power it is 50°F. The reactor conditions at which steam generator safety valves open is shown as a dashed line on Figure TS.2.1-1.

Except for special tests, POWER OPERATION with only one loop or with natural circulation is not allowed. SAFETY LIMITS for such special tests will be determined as a part of the test procedure.

The curves are based on the following nuclear hot channel factors (Reference 2):

$$F^N_{\Delta H} = 1.58\ [1 + 0.2(1-P)]\ ;\ \text{and}\ F^N_Q = 2.71$$

Use of these factors results in more conservative SAFETY LIMITS than would result from power distribution limits in Specification TS.3.10. Local peaking factors due to fuel densification are included in the hot channel factors, (Reference 4).

This combination of hot channel factors is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified by Figure TS.3.10-1 assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30 (Reference 3).

References

1. USAR, Section 3.2.1
2. USAR, Section 3.3.2
3. USAR, Section 14.4
4. WCAP 8091

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Bases

The reactor coolant system (Reference 1) serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the reactor coolant system is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system is assured. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III is 110% of design pressure.

The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure) has been established (Reference 2).

The nominal settings of the power-operated relief valves (2335 psig), the reactor high pressure trip (2385 psig) and the safety valves (2485 psig) have been established to assure that the pressure never reaches the reactor coolant system pressure SAFETY LIMIT.

In addition, the reactor coolant system safety valves (Reference 3) are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

The nominal settings of the five safety valves on each of the main steam lines are 1075 psig, 1090 psig, 1105 psig, 1120 psig, and 1129 psig.

As an assurance of system integrity, the reactor coolant system was hydrotested at 3107 psig prior to initial operation (Reference 4).

References

1. USAR, Section 4.1
2. USAR, Section 4.1.3.1
3. USAR, Section 4.4.3.2
4. USAR, Section 4.1

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases

The power range high flux reactor trips (low set point) provides redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis (Reference 1).

The intermediate and source range high flux reactor trips provide additional protection against uncontrolled startup excursions. As power level increases, during startup, these trips are manually blocked to prevent unnecessary plant trips.

The power range high flux (high set point) reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis (Reference 2).

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident (Reference 3).

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds) (Reference 4), and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors (Reference 2), is always below the core SAFETY LIMITS shown on Figure TS.2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced (References 5, 6).

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors (Reference 2).

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases continued

The overpower and overtemperature protection setpoints include the effects of fuel densification on core SAFETY LIMITS.

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis (Reference 7). The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency set point, ≥58.2 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error (Reference 2) and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system (Reference 8).

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of RATED THERMAL POWER.

The other reactor trips specified in 2.3.A.3. above provide additional protection. The trip initiated by steam/feedwater flow mismatch in coincidence with low steam generator water level is designed for protection from a sudden loss of the reactor's heat sink. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of-load transient.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases continued

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip compliments the power range nuclear flux high and low trip to assure that the criteria are met for rod ejection from partial power.

The negative power range rate trip provides protection against DNB for control rod drop accidents. Most rod drop events will cause a sufficiently rapid decrease in power to trip the reactor on the negative power range rate trip signal. Any rod drops which do not insert enough reactivity to cause a trip are analyzed to ensure that the core does not experience DNB. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

References

1. USAR, Section 14.4.1
2. USAR, Section 14.1
3. USAR, Section 14.6.1
4. USAR, Section 14.4.1
5. USAR, Section 7.4.1.1, 7.2
6. USAR, Section 3.3.2
7. USAR, Section 14.4.8
8. USAR, Section 14.1.10

BASES FOR

SECTION 3.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

3.0 Applicability

Bases

The intent of action statements which direct the plant to go to "at least HOT SHUTDOWN" is:

1. in POWER OPERATIONS the plant shall be placed in HOT SHUTDOWN,

2. in STARTUP OPERATIONS any plant heatup shall be stopped and the reactor coolant system boron concentration shall be at the concentration required to assure 1% shutdown margin at 200°F,

3. in any other condition above COLD SHUTDOWN, no heatup shall be allowed and the reactor coolant system boron concentration shall be at the concentration required to assure 1% shutdown margin at 200°F.

If the plant is required to be in some condition within 6 hours, and this condition is obtained within a shorter period of time, the time saved may be added to any required time to achieve another condition.

For example consider the following action statement: One of two pumps may be inoperable for 72 hours. If operability is not restored, be in HOT SHUTDOWN within the next 6 hours. If operability is not restored after an additional 48 hours, be in COLD SHUTDOWN within the following 30 hours. The action statement provides up to 156 hours (72 + 6 + 48 + 30) to achieve COLD SHUTDOWN.

1. If one of the pumps is discovered inoperable while in POWER OPERATIONS, and the plant was placed in HOT SHUTDOWN after 60 hours, COLD SHUTDOWN would need to be achieved within the next 96 hours (72 - 60 = 12, 12 + 6 + 48 + 30 = 96).

2. However, if this condition were discovered while in HOT SHUTDOWN, the unit could remain in HOT SHUTDOWN for the next 120 hours (72 + 48). The 6 hours for achieving HOT SHUTDOWN could not be used since the plant is already in that condition.

3. If this condition were discovered in between HOT SHUTDOWN and COLD SHUTDOWN, the plant could remain in that condition for 72 hours, and then be in COLD SHUTDOWN within the following 30 hours. The 6 hours for achieving HOT SHUTDOWN and the 48 hours for remaining at HOT SHUTDOWN could not be used since the plant is already below that condition.

The above paragraphs apply to all Section 3 requirements.

3.1 REACTOR COOLANT SYSTEM

Bases

A. Operational Components

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth greater than or equal to 50% of the 0.050-inch tube wall thickness as being unacceptable for POWER OPERATION. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents (Reference 2).

Part A of the specification requires that both reactor coolant pumps be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power PHYSICS TESTS, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

Reactor coolant pump start is restricted to RCS conditions where there is pressurizer level indication or low differential temperature across the SG tubes to reduce the probability of positive pressure surges causing overpressurization.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load (Reference 1).

The requirement that two groups of pressurizer heaters be OPERABLE provides assurance that at least one group will be available during a loss of offsite power to maintain natural circulation. Backup heater group "A" is normally supplied by one safeguards bus. Backup heater group "B" can be manually transferred within minutes to the redundant safeguards bus. Tests have confirmed the ability of either group to maintain natural circulation conditions.

The pressurizer power operated relief valves (PORVs) operate to relieve reactor coolant system pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The PORVs are pneumatic valves operated by instrument air. They fail closed on loss of air or loss of power to their DC solenoid valves. The PORV block valves are motor operated valves supplied by the 480 volt safeguards buses.

With RCS temperatures less than the MINIMUM PRESSURIZATION TEMPERATURE (MPT), the RCS safety valves and normal setpoints on the pressurizer PORVs do not provide overpressure protection for certain operational transients. The low temperature overpressure mitigating system installed at Prairie Island is designed to prevent pressurizing the RCS above the pressure limits specified in Figures TS.3.1-1 and TS.3.1-2 (Reference 3). OPERABILITY of an overpressure mitigating system PORV requires that the low pressure set point has been selected (enabled), the upstream isolation valve is open and the backup air supply is charged.

The system is designed to perform its function in the event of a single failure and is designed to meet the requirements of IEEE-279. The backup air supply provides sufficient air to operate the PORVs following a letdown isolation with one charging pump in operation for a period of ten minutes after receipt of the overpressure alarm. These specifications provide assurance that the overpressure mitigating system will perform its intended function.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

The Specifications require that at least two methods of removing decay heat are available for each reactor. Above 350°F, both steam generators must be operable to serve this function. Below 350°F, either a steam generator (and RCP) or a residual heat removal loop are capable of removing decay heat and any combination of two loops is specified. If redundant means are not available, the reactor is placed in COLD SHUTDOWN.

The reactor coolant vent system is provided to exhaust noncondensible gases from the reactor coolant system that could inhibit natural circulation core cooling. The OPERABILITY of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable. An inoperable vent path valve is defined as a valve which cannot be opened or whose position is unknown.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow of the reactor coolant makeup system.

References

1. USAR, Section 14.4.8.

2. Testimony by J Knight in the Prairie Island Public Hearing on January 28, 1975.

3. NSP Letter to USNRC, "Reactor Vessel Overpressurization", dated July 22, 1977.

3.1 REACTOR COOLANT SYSTEM

Bases continued

B. Pressure/Temperature Limits

The reactor coolant system heatup and cooldown rates in Figures TS.3.1-1 and TS.3.1-2 are applicable to both Unit 1 and Unit 2. The curves are based on Unit 1 toughness data and are conservative for the Unit 2 vessel. Toughness data is included in Tables TS.3.1-1 and TS.3.1-2.

The reactor coolant temperature/pressure and system heatup/cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS.3.1-1 and TS.3.1-2 for the first full-power service period.

1. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

2. Figures TS.3.1-1 and TS.3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

The limit lines shown in Figures TS.3.1-1, TS.3.1-2 shall be recalculated periodically using methods discussed below.

The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

The criticality limit specified in Figure TS.3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters. The pressurizer heater and associated power cables have been sized for continuous operation at full heater power.

3.1 REACTOR COOLANT SYSTEM

Bases continued

B. Pressure/Temperature Limits (continued)

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1972 Summer Addenda, (Reference 1). Heatup and cooldown limit curves are calculated using the most limit curves are calculated using the most limiting value of RT_{NDT} determined as follows:

1. Determine the highest RT_{NDT} of the material in the core region of the reactor vessel using original values from Tables TS.3.1-1 and TS.3.1-2 and estimating the radiation induced ΔRT_{NDT} using Figure TS.3.1-3. Fast neutron (above 1 Mev) fluence at the 1/4 T and 3/4 T vessel locations are given as a function of full power service life in Figure TS.3.1-4.

2. Examine the data for all other ferritic materials in the reactor coolant pressure boundary to assure that no other component will be limiting.

3. Initially, the effect of radiation of the RT_{NDT} of the reactor vessel core region material is estimated using the curves shown in Figure TS.3.1-3. The ΔRT_{NDT} shown for the first full power service period is factored into the heatup and cooldown curves provided. Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, when evaluated according to ASTM E185, indicate that they are inappropriate. At this time, the heatup and cooldown curves must be recalculated.

The length of the first full power service period has been chosen such that the limiting RT_{NDT} at the 1/4 T vessel location has a radiation induced ΔRT_{NDT} on the order of 100-150°F. The assumption of a radiation induced shift of this magnitude assures that all other components in the primary pressure boundary will be operated conservatively, in accordance with code recommendations (Reference 2).

Heatup and Cooldown Limit Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G2000 in Section III of the ASME Boiler and Pressure Vessel Code; and discussed in detail in (Reference 2).

3.1 REACTOR COOLANT SYSTEM

Bases continued

B. Pressure/Temperature Limits (continued)

The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve (Reference 1) for the metal temperature at that time. Furthermore, the approach applies explicit safety factors of 2.0 and 1.25* on stress intensity factors induced by pressure and thermal gradients, respectively. Thus, the governing equation for the heatup-cooldown analysis is:

$$2K_{Im} + 1.25\,K_{It} \leq K_{IR} \qquad (1)$$

where:

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce

* The 1.25 safety factor on K_{It} represents additional conservatism above Code requirements.

3.1 REACTOR COOLANT SYSTEM

Bases continued

B. Pressure/Temperature Limits (continued)

stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses as 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the ID flow is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rated than for steady state under certain conditions.

3.1 REACTOR COOLANT SYSTEM

Bases continued

B. Pressure/Temperature Limits (continued)

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure TS.3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details for these calculations are provided in (Reference 2).

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed i accordance with Code requirements.

References

1. ASME Boiler and Pressure Vessel Code, Section III, 1972 Summer Addenda, Appendix G.

2. WCAP-7924. W.S. Hazelton, "Basis for Heatup and Cooldown Limit Curves", WCAP-7924, June 1972.

3.1 REACTOR COOLANT SYSTEM

Bases continued

C. Reactor Coolant System Leakage

Leakage from the reactor coolant system is collected in the containment or by other systems. These systems are the main steam system condensate and feedwater system, and the chemical and volume control system.

Detection of leaks from the reactor coolant system is by one or more of the following (Reference 1):

1. An increased amount of makeup water required to maintain normal level in the pressurizer.

2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.

3. Containment sump water level indication.

4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitors, the area radiation monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor (Reference 2).

A leak rate of 1 gpm corresponds to a through wall crack less than 0.6 inches long bases on test data. Steam generator tubes having a 0.6-inch long through-wall crack have been shown to resist failure at pressures resulting from normal operation, LOCA, or steam line break accidents (Reference 3).

Specification 3.1.C.3 provides for actions to be taken in the event of failure or excessive leakage of a check valve which isolates the high pressure reactor coolant system from the low pressure RHR system piping.

References

1. USAR, Section 6.5
2. USAR, Section 7.5.1
3. Testimony by J Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

3.1 REACTOR COOLANT SYSTEM

Bases continued

D. Maximum Coolant Activity

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Prairie Island site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

Specification 3.1.D.2, permitting power operation to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the allowable limit shown on Figure TS.3.1-5, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure TS.3.1-5 should be minimized since the activity levels allowed by Figure TS.3.1-5 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing RCS temperature to less than 500°F prevents the release of activity should a steam generator tupe rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements in Table TS.4.1-2B provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

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 concentration 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

3.1 REACTOR COOLANT SYSTEM

Bases continued

F. Reactor Coolant Temperature Conditions for Criticality

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. The safety analyses conducted for Prairie Island units with Westinghouse fuel assumed a non-positive moderator temperature coefficient. The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during startup physics testing, whereas the moderator temperature coefficient is an inferred parameter determined by subtracting the predicted fuel temperature coefficient from the experimentally determined isothermal temperature coefficient.

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive moderator temperature coefficient could exist at beginning of cycle (BOC). For cycles with Exxon fuel, safety analyses are conducted assuming a positive moderator temperature coefficient. These analyses predict the isothermal coefficient to be negative at an all rods out, hot zero power condition. Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F.1 requirements are waived during low power physics tests to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these physics tests. In addition, the strong negative Doppler coefficient[1] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

References:
(1) FSAR Figure 3.2.10

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Bases

The chemical and volume control system provides control of the reactor coolant system boron inventory (Reference 1). This is normally accomplished by using any one of the three charging pumps in series with any one of the four boric acid pumps. The design of the two-unit plant permits the alignment of any of the four boric acid transfer pumps to either reactor. An alternate method of boration will be use of the charging pumps taking suction directly from the refueling water storage tank. A third method will be to use the safety injection pumps. There are two sources of borated water available for injection to the core through 3 different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents to the suction of the charging pumps that can inject it to the reactor coolant system through the charging line or the reactor coolant pump seals.

(2) The charging pumps can take suction directly from the refueling water storage tank. (1950 ppm boron solution. Reference is made to Specification 3.3.A.1.a.)

(3) The safety injection pumps can take their suctions from either the boric acid tanks or the refueling water storage tank and inject the contents to the reactor coolant system through the high head safety injection piping.

The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach COLD SHUTDOWN at any time during core life.

Approximately 1800 gallons of at least 11.5% solution of boric acid are required to meet COLD SHUTDOWN condition. Thus, a minimum of 2000 gallons in the boric acid tank is specified. If the safety injection system is actuated while there are only 2000 gallons in the boric acid tank, more than 600 gallons of concentrated boric acid solution would be injected into the core before the pump suction is transferred to the refueling water storage tank. This 600 gallons injected into the core is more than sufficient to counteract the effects of the rupture of a steam pipe (Reference 2).

In order to ensure solution solubility at the boric acid concentration in the system, a minimum temperature of 145°F is required. Two channels of heat tracing are installed on lines normally containing concentrated boric solution to maintain the required minimum temperature.

References

1. USAR Section 10.2.3.2
2. USAR Section 14.5.5

3.3 ENGINEERED SAFETY FEATURES

Bases

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant (Reference 1). With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during POWER OPERATION and therefore, to be conservative, most engineered safety system components and auxiliary cooling systems shall be fully OPERABLE. During low temperature PHYSICS TEST there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safeguard systems are not required.

The OPERABLE status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full OPERABILITY within a relatively short time. Limited conditions of permissible temporary outage of redundant components are prescribed for specific time intervals that are consistent with minor maintenance. These permissible conditions and time intervals are specified in such a manner as to apply identically during sustained POWER OPERATION and during recovery from inadvertent scram or from shutdown compelled by Technical Specifications because of outage of a specific component that has again been made OPERABLE. The transient condition of restart in the latter cases in no way alters the types of safeguards equipment nor the extent of redundancy that must be available.

Inoperability of a single component does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. If it develops that the inoperable component is not repaired within the specified allowable time period or that a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. If the malfunction(s) are not corrected,

3.3 ENGINEERED SAFETY FEATURES

Bases continued

after 48 hours in HOT SHUTDOWN, the reactor will be placed in the COLD SHUTDOWN condition, within the next 30 hours. In the COLD SHUTDOWN condition there is no possibility of a LOCA that would release fission products or damage the fuel elements.

The specified intervals for equipment inoperability are based on:

(1) Assuring with high reliability that the safety system will function properly if required to do so.

(2) Allowance of sufficient time to complete required repairs and testing using safe and proper procedures.

Assuming the reactor has been operating at full RATED POWER for at least 100 days, the magnitude of the decay heat decreases as follows after initiating HOT SHUTDOWN.

Time After Shutdown	Decay Heat, % of RATED POWER
1 min.	4.5
30 min.	2.0
1 hour	1.62
8 hours	0.96
48 hours	0.62

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during POWER OPERATION. Putting the reactor in the HOT SHUTDOWN condition significantly reduced the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs. Failure to complete repairs within 48 hours after placing the reactor in the HOT SHUTDOWN condition is considered indicative of need for major maintenance, and in such cases the reactor would therefore be placed in the COLD SHUTDOWN condition.

The accumulator and refueling water tank conditions specified are consistent with those assumed in the LOCA analysis (Reference 2).

The containment cooling function is provided by two independent systems: containment fan cooler units and containment sprays. During normal operation, four containment fan cooler units are utilized to remove heat lost from equipment and piping within the containment. In the event of the Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: four containment fan cooler units, two containment spray pumps, or two containment fan cooler units plus one containment spray pump (Reference 4). Two of the four containment fan cooler units are permitted to be

3.3 ENGINEERED SAFETY FEATURES

Bases continued

inoperable during POWER OPERATION. This is an abnormal operating situation, in that plant operating procedures require that inoperable containment fan cooler units be repaired as soon as practical. However, because of the difficulty of access to make repairs, it is important on occasion to be able to operate temporarily with only two containment fan cooler units. Two containment fan cooler units can provide adequate cooling for normal operation when the containment fan cooler units are cooled by the chilled water system (Reference 3). Compensation for this mode of operation is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit, either following a loss-of-coolant accident or during normal plant shutdown. The four pumps of the two-unit facility can be cross-connected as necessary to accommodate temporary outage of the pump. If, during the post-accident phase, the component cooling water supply were lost, core and containment cooling could be maintained until repairs were effected (Reference 5).

Normal cooling water supply is from two motor-driven pumps backed up by a third motor-driven pump (Reference 6). In the event of complete loss of station power, cooling water is supplied by two diesel-driven pumps which start automatically, each serving half the fan coolers in each reactor. Operation of a single cooling water pump of either type provides sufficient cooling in one unit during the injection and recirculation phases of a postulated loss-of-coolant accident plus sufficient cooling to maintain the second unit in a hot standby condition.

The component cooling water system and the cooling water system provide water for cooling components used in normal operation, such as turbine generator components, and reactor auxiliary components in addition to supplying water for accident functions. These systems are designed to automatically provide two separate redundant paths in each system following an accident. Each redundant path is capable of cooling required components in the unit having the accident and in the operating unit.

There are several manual valves and manually-controlled motor-operated valves in the engineered safety feature systems that could, if one valve is improperly positioned, prevent the required injection of emergency coolant (Reference 7). These valves are used only when the reactor is subcritical and there is adequate time for actuation by the reactor operator. To ensure that the manual valve alignment is appropriate for safety injection during power operation, these valves are tagged and the valve position will be changed only under direct

3.3 ENGINEERED SAFETY FEATURES

Bases continued

administrative control. For the motor-operated valves, the motor control center supply breaker is physically locked in the open position to ensure that a single failure in the actuation circuit or power supply would not move the valve.

Specification 3.3.A.3 allows use of an SI pump to perform operations required at low RCS temperatures; e.g., raising accumulator levels in order to meet the level requirement of Specification 3.3.A.1.b(2) or ASME Section XI tests of the SI system check valves.

Specification 3.3.A.3 also allows use of both SI pumps at low temperatures for conduct of the integrated SI test. In this case, pressurizer level is maintained at less than 50% and the SI pump discharge valves are shut to prevent fluid injection into the RCS. This combination of conditions under strict administrative control assure that overpressurization cannot occur. The option of having the reactor vessel head removed is allowed since in this case RCS overpressurization cannot occur.

References

1. USAR, Section 3.3.2
2. USAR, Section 14.6.1
3. USAR, Section 6.3.2
4. USAR, Section 6.3
5. USAR, Section 10.4.2
6. USAR, Section 10.4.1
7. USAR, Figure 6.2-1
   USAR, Figure 6.2-2
   USAR, Figure 6.2-5
   USAR, Figure 10.2-11

3.4 STEAM AND POWER CONVERSION SYSTEMS

Bases

A reactor shutdown from power requires removal of decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten main steam safety valves have a total combined rated capability of 7,745,000 lbs/hr. The total full power steam flow is 7,094,000 lbs/hr; therefore, the ten main steam safety valves will be able to relieve the total steam flow if necessary (Reference 1).

In the unlikely event of complete loss of offsite electrical power to either or both reactors, continued removal of decay heat would be assured by availability of either the steam-driven auxiliary feedwater pump or the motor-driven auxiliary feedwater pump associated with each reactor, and by steam discharge to the atmosphere through the main steam safety valves. One auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from one reactor. The motor-driven auxiliary feedwater pump for each reactor can be made available to the other reactor.

The minimum amount of water specified for the condensate storage tanks is sufficient to remove the decay heat generated by one reactor in the first 24 hours of shutdown. Essentially unlimited replenishment of the condensate storage supply is available from the intake structures through the cooling water system.

The two power-operated relief valves located upstream of the main steam isolation valves are required to remove decay heat and cool the reactor down following a high energy line rupture outside containment (Reference 2). Isolation dampers are required in ventilation ducts that penetrate those rooms containing equipment needed for the accident.

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

Reference

1. USAR, Section 11.9.4
2. FSAR, Appendix I

3.5 INSTRUMENTATION SYSTEM

Bases

Instrumentation has been provided to sense accident conditions and to initiate reactor trip and operation of the Engineered Safety Features (Reference 1).

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss-of-coolant accident (LOCA) is provided by a high containment pressure signal backed up by the low pressurizer pressure signal. These conditions would accompany the depressurization and coolant loss during a LOCA.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the low pressurizer pressure signal and, in case of a vreak within the containment, by the high containment pressure signal.

The safety injecting of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

Containment Spray

Containment sprays are also actuated by a high containment pressure signal (Hi-Hi) to reduce containment pressure in the event of a loss-of-coolant or steam line break accident inside the containment.

The containment sprays are actuated at a higher containment pressure (approximately 50% of design containment pressure) than is safety injection (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated on coincidence of high containment pressure sensed by three sets of one-out-of-two containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the environment in the event of a loss-of-coolant accident.

Steam Line Isolation

In the event of a steam line break, the steam line stop valve of the affected line is automatically isolated to prevent continuous, uncontrolled steam release from more than one steam generator. The steam lines are isolated on high containment pressure (Hi-Hi) or high steam

3.5 INSTRUMENTATION SYSTEM

Bases continued

Steam Line Isolation (continued)

line flow in coincidence with low T_{avg} and safety injection or high steam flow (Hi-Hi) in coincidence with safety injection. Adequate protection is afforded for breaks inside or outside the containment even when it is assumed that the steam line check valves do not function properly.

Containment Ventilation Isolation

Valves in the containment purge and inservice purge systems automatically close on receipt of a Safety Injection signal or a high radiation signal. Gaseous and particulate monitors in the exhaust stream or a gaseous monitor in the exhaust stack provide the high radiation signal.

Ventilation System Isolation

In the event of a high energy line rupture outside of containment, redundant isolation dampers in certain ventilation ducts are closed (Reference 4).

Safeguards Bus Voltage

Relays are provided on buses 15, 16, 25, and 26 to detect loss of voltage and degraded voltage (the voltage level at which safety related equipment may not operate properly). On loss of voltage, the automatic voltage restoring scheme is initiated immediately. When degraded voltage is sensed, the voltage restoring scheme is initiated if acceptable voltage is not restored within a short time period. This time delay prevents initiation of the voltage restoring scheme when large loads are started and bus voltage momentarily dips below the degraded voltage setpoint.

Auxiliary Feedwater System Actuation

The following signals automatically start the pumps and open the steam admission control valve to the turbine driven pump of the affected unit:

1. Low-low water level in either steam generator
2. Trip of both main feedwater pumps
3. Safety Injection signal
4. Undervoltage on both 4.16 kV normal buses (turbine driven pump only)

Manual control from both the control room and the Hot Shutdown Panel are also available. The design provides assurance that water can be supplied to the steam generators for decay heat removal when the normal feedwater system is not available.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Limiting Instrument Setpoints

1. The high containment pressure limit is set at about 10% of the maximum internal pressure. Initiation of Safety injection protects against loss of coolant (Reference 2) or steam line break accidents as discussed in the safety analysis.

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant (Reference 2) or steam line break accidents (Reference 3) as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis (Reference 2).

4. The steam line low pressure signal is lead/lag compensated and its setpoint is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis (Reference 3).

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full-load flow at the full load pressure in order to protect against large steam break accidents. The coincident low T_{avg} setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam break (Reference 3).

6. Steam generator low-low water level and 4.16 kV Bus 11 and 12 (21 and 22 in Unit 2) low bus voltage provide initiation signals for the Auxiliary Feedwater System. Selection of these setpoints is discussed in the Bases of Section 2.3 of the Technical Specification.

7. High radiation signals providing input to the Containment Ventilation Isolation circuitry are set in accordance with the Radioactive Effluent Technical Specifications. The setpoints are established to prevent exceeding the limits of 10 CFR Part 20 at the SITE BOUNDARY.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Limiting Instrument Setpoints (continued)

8. The degraded voltage protection setpoint is 90± 2% of nominal 4160 V bus voltage. Testing and analysis have shown that all safeguards loads will operate properly at or above the degraded voltage setpoint. The degraded voltage protection time delay of 6±2 seconds has been shown by testing and analysis to be long enough to allow for voltage dips resulting from the starting of large loads. This time delay is also consistent with the maximum time delay assumed in the ECCS analysis for starting of a safety injection pump. A maximum limit on the degraded voltage setpoint has been established to prevent unnecessary actuation of the voltage restoring scheme.

   The loss of voltage protection setpoint is approximately 55% of nominal 4160 V bus voltage. Relays initiate a rapid (less than two seconds) transfer to an alternate source on loss of voltage.

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for CHANNEL CALIBRATION and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in a concurrent channel.

References

1. USAR, Section 7.4.2
2. USAR, Section 14.6.1
3. USAR, Section 14.5.5
4. FSAR, Appendix I

3.6 CONTAINMENT SYSTEM

Bases

Proper functioning of the Shield Building vent system is essential to the performance of the containment system. Therefore, except for reasonable periods of maintenance outage for one redundant chain of equipment, the system should be wholly in readiness whenever above 200°F. Proper functioning of the auxiliary building special vent system and isolation of the auxiliary building normal vent system are similarly necessary to preclude possible unfiltered leakage through penetrations that enter the special ventilation zone.

The auxiliary building special ventilation zone and its associated ventilation system have been designed to serve as secondary containment following a loss of coolant accident (Reference 2). Special care was taken to design the access doors in the boundary and isolation valves in normal ventilation systems so that AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY can be intact during reactor operation. The zone can perform its accident function with openings if they can be closed within 6 minutes, since the accident analysis assumed direct leakage of primary containment atmosphere to the environs when the shield building is at positive pressure (6 minutes). As noted in Reference 2, part of the Shield Building is part of the Auxiliary Building Special Ventilation Zone Integrity. The part of the Shield Building which is part of the Auxiliary Building Special Ventilation Zone is subject to the Technical Specifications of the Shield Building Integrity and not those associated with Auxiliary Building Special Ventilation Zone Integrity.

The action statement which allows Shield Building Integrity to be lost for 24 hours will allow for modifications to be made to the Shield Building during power operations.

The COLD SHUTDOWN condition precludes any energy release or buildup of containment pressure from flashing of reactor coolant in the event of a system break.

The shutdown margin for the COLD SHUTDOWN condition assures sub-criticality with the vessel closed, even if the most reactive rod control cluster assembly were inadvertently withdrawn.

The 2 psig limit on internal pressure provides adequate margin between the maximum internal pressure of 46 psig and the peak accident pressure resulting from the postulated Design Basis Accident (Reference 1).

The containment vessel is designed for 0.8 psi internal vacuum, the occurrence of which will be prevented by redundant vacuum breaker systems.

The containment has a nil ductility transition temperature of 0°F. Specifying a minimum temperature of 30°F will provide adequate margin above NDTT during power operation when containment is required.

The conservative calculation of off-site doses for the loss of coolant accident (References 2, 4) is based on an initial shield building annulus air temperature of 60°F and an initial containment vessel air tempera-

3.6 CONTAINMENT SYSTEM

Bases continued

ture of 104°F. The calculated period following LOCA for which the shield building annulus pressure is positive, and the calculated off-site doses are sensitive to this initial air temperature difference. The specified 44°F temperature difference is consistent with the LOCA accident analysis (Reference 4).

The initial testing of inleakage into the shield building and the auxiliary building special ventilation zone (ABSVZ) has resulted in greater specified inleakage (Figure TS.4.4-1, change No. 1) and the necessity to deenergize the turbine building exhaust fans in order to achieve a negative pressure in the ABSVZ (TS.3.6.E.2). The staff's conservative calculation of doses for these conditions indicated that changing allowable containment leak rate from 0.5% to 0.25%/day would offset the increased leakage (Reference 3).

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The Charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

References

1. USAR, Section 5
2. USAR, Section 10.3.4 and FSAR Appendix G
3. Letter to NSP dated November 29, 1973
4. Letter to NSP dated September 16, 1974

3.7 AUXILIARY ELECTRICAL SYSTEM

Bases

The intent of this specification is to provide assurance that at least one external source and one standby source of electrical power is always available to accomplish safe shutdown and containment isolation and to operate required engineered safeguards equipment following an accident.

Plant auxiliary power can be supplied from four separate external power sources which have multiple off-site network connections: the reserve transformer from the 161 kV portion of the plant substation; the second reserve transformer from the 345 kV portion of the plant substation and the two cooling tower transformers, one of which is supplied from a tertiary winding on the substation auto transformer, and the other directly from the 345 kV switchyard. Any one of the four sources is sufficient, under analyzed conditions, to supply all the necessary accident and post-accident load requirements for one reactor, along with the shutdown of the second reactor.

Each source separately supplies the safeguards buses in such manner that items of equipment which are redundant to each other are supplied by separate sources and buses.

Each diesel generator is connected to one 4160 volt safeguards bus in each of the two reactors and has sufficient capacity to start sequentially and operate the safeguards equipment supplied by one bus. The set of safeguards equipment items supplied by each bus is, alone, sufficient to maintain adequate cooling of the fuel and to maintain containment pressure within the design value in the event of a loss-of-coolant accident.

Each diesel starts automatically upon low voltage on its associated bus in either unit and both diesel generators start in the event of a safety injection signal for either reactor. The minimum fuel supply of 70,000 gallons will supply one diesel cooling water pump and one diesel generator (loaded per USAR Table 8.4-1) for greater than 14 days. Additional diesel fuel can normally be obtained within a few hours. This assures an adequate supply even in the event of the probable maximum flood.

Following the inoperability of a Diesel Generator, the other diesel generator is tested to prove that the cause of the inoperability does not affect both diesel generators. However, if the diesel generator is inoperable due to preplanned preventative maintenance, operability of the other diesel generator does not need to be proven.

3.7 AUXILIARY ELECTRICAL SYSTEM

Bases continued

The plant 125 volt d-c power is normally supplied by two batteries for each plant, each of which will have a battery charger in service to maintain full charge and to assure adequate power for starting the diesel generators and supplying other emergency loads.

The arrangement of the auxiliary power sources and equipment and this specification assure that no single fault condition will deactivate more than one redundant set of safeguard equipment items in one reactor and will therefore not result in failure of the plant protection system to respond adequately to a loss-of-coolant accident.

Reference

USAR, Section 8
USAR, Figure 8.2-2

3.8 REFUELING AND FUEL HANDLING

Bases

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the precautions specified above, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during CORE ALTERATIONS that would result in a hazard to public health and safety (Reference 1). Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

Under rodded and unrodded conditions, the K_{eff} of the reactor must be less than or equal to 0.95 and the boron concentration must be greater than or equal to 2000 ppm as indicated in 3.8.A.4. Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. 3.8.A.9 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

No movement of fuel in the reactor is permitted until the reactor has been subcritical for at least 100 hours to permit decay of the fission products in the fuel. The delay time is consistent with the fuel handling accident analysis (Reference 2).

The spent fuel assemblies will be loaded into the spent fuel cask after sufficient decay of fission products. While inserting and withdrawing the cask into pool No. 1, the cask will be suspended above the bottom of the pool up to a maximum of 42 feet. The consequences of potential load drops have been evaluated in accordance with NUREG-0612 (Reference 4). Following is a discussion of the basis for the limitations which resulted from that evaluation.

The cask will not be inserted into the pool until all fuel stored in the pool has been discharged from the reactor a minimum of 5 years. Supporting analysis indicated that fuel stored in the pool for a period as short as 50 days would allow sufficient decay of the fission products such that their release would result in off-site doses less than 25% of the 10CFR Part 100 guidelines. The five year decay period was selected in following the general principle that spent fuel with the longest decay time would result in the least off-site doses in the event of an accident, while providing the plant operational flexibility. The cask will not be inserted or withdrawn from the pool unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that if fuel is crushed by a cask

3.8 REFUELING AND FUEL HANDLING

Bases continued

drop, k_{eff} will be less than or equal to 0.95. The cask will not be inserted or withdrawn from the pool unless a cask impact limiter, crash pad, or combination thereof is in place with the capability to absorb energy of a cask drop such that no significant amount of water leakage results from pool structural damage. This is to ensure that at no time will water level drop below the top of the spent fuel stored in the pool. In loading the cask into a carrier, there is a potential drop of 66 feet (Reference 3). The cask will not be loaded onto the carrier for shipment prior to a 3-month storage period. At this time, the radioactivity has decayed so that a release of fission products from all fuel assemblies in the cask would result in off-site doses less than 10 CFR Part 100. It is assumed, for this dose analysis that 12 assemblies rupture after storage for 90 days. Other assumptions are the same as those used in the dropped fuel assembly accident in the SER, Section 15. The resultant doses at the SITE BOUNDARY are 94 Rems to the thyroid and 1 Rem whole body.

The number of recently discharged assemblies in Pool No. 1 has been limited to 45 to provide assurance that in the event of loss of pool cooling capability, at least eight hours are available under worst case conditions to make repairs until the onset of boiling.

The Spent Fuel Pool Special Ventilation System (Reference 3) is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation system is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

During movement of irradiated fuel assemblies or control rods, a water level of 23 feet is maintained to provide sufficient shielding.

3.8 REFUELING AND FUEL HANDLING

Bases continued

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The water level may also be lowered below 20 feet for upper internals removal/replacement. The basis for these allowance(s) are (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching/unlatching and upper internals removal/replacement the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

References

1. USAR, Section 10.2.1.2
2. USAR, Section 14.5.1
3. USAR, Section 10.3
4. USAR, Section 10.3.4.2.1
Exhibit C, NSP License Amendment Request Dated December 21, 1984.

3.9 RADIATION EFFLUENT

Bases

A. Liquid Effluents

Specification 3.9.A.1 is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures exceeding (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

Specification 3.9.A.2.a is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The LIMITING COMDITION FOR OPERATION implements the guides set forth in Section II.A of Appendix I. Action required by Specification 3.8.A.2.B provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". Considering that the nearest drinking water supply using the river for drinking water is more than 300 miles downstream, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the drinking water that are in excess of the requirements of 40 CFR 141.

Specification 3.9.A.3 provides assurance that the liquid radwaste treatment system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as lo as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

The liquid radwaste treatment system is shared by both units. It is not practical to determine the contribution from each unit to liquid radwaste releases. For this reason, liquid radwaste releases will be allocated equally to each unit.

3.9 RADIATION EFFLUENT

Bases continued

A. Liquid Effluents (continued)

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the contents of the tank, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, in an UNRESTRICTED AREA.

B. Gaseous Effluents

Specification 3.9.B.1.a is provided to ensure that the dose rate at any time at the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either inside or outside the site boundary to annual average concentrations exceeding limits specified in Appendix B, Table II of 10 CFR Part 20 [10 CFR Part 20.106(b)]. For individuals who may at times be within the SITE BOUNDARY, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to less than or equal to 1500 mrem/year at the SITE BOUNDARY.

Specification 3.9.B.2.a is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The LIMITING CONDITIONS FOR OPERATION implement the guides set forth in Section II.B of Appendix I. Action required by Specification 3.9.B.2.b provides the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable".

Specification 3.9.B.3.a is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The LIMITING CONDITIONS FOR OPERATION are the guides set forth in Section II.C of Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". The release rate specifications for I-131, tritium and radioactive particulates with half-lives greater than eight days are dependent on the existing radionuclide pathways to man in the UNRESTRICTED AREA. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation

3.9 RADIATION EFFLUENT

Bases continued

B. Gaseous Effluents (continued)

with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

Specification 3.9.B.4.a provides assurance that the Waste Gas Treatment System and the Ventilation Exhaust Treatment Systems will be available for use whenever gaseous wastes are released to the environment. The requirement that the appropriate portions of the Waste Gas Treatment System be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

Specification 3.9.B.4.c, 3.9.B.4.d, and 3.9.B.4.e are provided to ensure the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. Maintaining the concentrations below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

The waste gas system is a pressurized system with two potential sources of oxygen: 1) oxygen added for recombiner operation, and 2) placing tanks vented for maintenance back on the system. The system is operated with flow through the recombiners and with excess hydrogen in the system. By verifying that oxygen is less than or equal to 2% at the recombiner outlet, there will be not explosive mixtures in the system. Waste gas system oxygen is monitored by the two recombiner oxygen analyzers and the 121 gas analyzer. The 121 gas analyzer only monitors the low level loop of the waste gas system. If the required gas analyzers are not operable, the oxygen to the recombiner will be isolated to prevent oxygen from entering the system from this source. Tanks that may undergo maintenance are normally purged with nitrogen before placing them in service to eliminate this as a source of oxygen.

Specification 3.9.B.4.f is provided to limit the radioactivity which can be stored in one decay tank. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that

3.9 RADIATION EFFLUENT

Bases continued

B. Gaseous Effluents (continued)

in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

Specification 3.9.B.5.a requires the containment to be PURGED, during reactor operation, through the inservice purge system. This provides for iodine and particulate removal from the purge release. During outages when the containment is opened for maintenance, the containment ventilation exhaust is directed to the monitored reactor building vent.

The cooling towers at Prairie Island are located to the south of the plant and are within the 50°-arc described in this specification. At low wind, velocities (below 10 mph) the gaseous activity released from the gaseous radwaste system could be at or near ground level near the cooling towers and remain long enough to be drawn into the circulating water in the tower. This specification minimizes the possibility of releases from the gaseous radwaste system from entering the river from tower scrubbing.

The 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. For this reason, it is not practical to allocate releases to any specific unit. All releases will be allocated equally in determining conformance to the design objectives of 10 CFR Part 50, Appendix I.

C. Solid Radioactive Waste

The OPERABILITY requirements placed on the solid radwaste system ensure that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/SOLIDIFICATION agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3.9 RADIATION EFFLUENT

Bases continued

D. Dose from all Uranium Fuel Cycle Sources

This specification is promised to meet the dose limitations of 40 CFR 190. This specification requires the preparation and submittal of a special report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. Submittal of the report is considered a timely request, and a variance is granted until NRC staff action on this request is complete. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a real individual will exceed 40 CFR 190 if the individual reactors remain within the reporting requirement level. For the purposes of the special report, it may be assumed that the dose commitment to the real individual from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

E. & F. Effluent Monitoring Instrumentation

These specifications are provided to assure that effluent release points are continuously monitored.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonymous.

A. Shutdown Reactivity

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

B. Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core of greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

During operation, the plant staff compares the measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, (described later) to the limits determined in the transient and LOCA analyses. The limiting $F_Q(Z)$ includes measurement, engineering, and calculational uncertainties. The terms on the right side of the equations in Section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. The maximum value of $F_Q(Z)$ is 2.32/P for the Prairie Island reactors. This value is restricted further by the K(Z) and $BU(E_j)$ functions described below. The product of these three factors is $F_Q^j(Z)$.

The K(Z) function shown in Figure TS.3.10-5 is a normalized function that limits $F_Q(Z)$ axially for three regions. The K(Z) specified for the lowest six (6) feet of the core is abritrarily flat since the lower part of the core is generally not limiting. Above that region, the K(Z) value is based on large and small break LOCA analyses. $F_Q(Z)$ in the uppermost region is limited to reduce the PCT expected during a small break LOCA since this region of the core is expected to uncover temporarily for some small break LOCAs.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

The $BU(E_i)$ function shown in Figure TS.3.10-7 is a normalized function that limits $F_Q(Z)$ based on exposure dependent analyses for the ENC fuel. These analyses consider pin internal pressure uncertainties, fuel swelling, rupture pressures and flow blockage.

F_Q^N is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux in the core divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

V(Z) is an axially dependent function applied to the equilibrium measured F_Q^N to bound F_Q^N's that could be measured at non-equilibrium conditions. This function is based on power distribution control analyses that evaluated the effect of burnable poisons, rod position, axial effects, and xenon worth.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local veriations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

The 1.05 multiplier accounts for uncertainties associated with measurement of the power distribution with the movable incore detectors and the use of those measurements to establish the assembly local power distribution.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizational (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

In the specified limit of $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N$ 1.55/1.08. The logic behind the larger uncertainty in this case is that:

(a) abnormal perturbations in the radial power shape (e.g., rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q^N,

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

(b) the operator has a direct influence on F_Q^N through movement of rods, and can limit it to the desired value, while he has no direct control over $F_{\Delta H}^N$ and,

(c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

When a measurement of $F_{\Delta H}^N$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least once each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An accidental misalignment limit of 13 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.

3. The control bank insertion limits are not violated.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial affect which is defined as the difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ and F^N_Q allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In specification 3.10, F^N_Q is arbitrarily limited for P less than or equal to 0.5 (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the $F_Q(Z)$ upper bound envelope of 2.32/P times Figures TS.3.10-5 and TS.3.10-7 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. Figure TS.3.10-6 shows the allowed deviation from the target flux difference as the function of THERMAL POWER.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

The alarms provided are derived from the plant process computer which determines the one minute averages of the operable excore detector outputs to monitor indicated axial flux difference in the reactor core and alerts the operator when indicated axial flux difference alarm conditions exist. Two types of alarm messages are output. Above a preset (90%) power level, an alarm message is output immediately upon determining a delta flux (as determined from two operable excore channels) exceeding a preset band about a target delta flux value. Below this preset power level, an alarm message is output if the indicated axial flux difference (as determined from two OPERABLE excore channels) exceeded its allowable limits for a preset cumulative (usually 1 hour) amount of time in the past 24 hours. For periods during which the alarm on flux difference in inoperable, manual surveillance will be utilized to provide adequate warning of significant variations in expected flux differences. However, every attempt should be made to restore the alarm to an operable condition as soon as possible. Any deviations from the target band during manual logging would be treated as deviations during the entire preceding logging interval and appropriate actions would be taken. This action is necessary to satisfy NRC requirements; however, more frequent readings may be logged to minimize the penalty associated with a deviation from the target band to justify continued operation at the current power. The time that deviations from the target band occur are normally accumulated by the computer above 15% power. Below 15% the probability of exceeding the allowable limits becomes increasingly smaller as it becomes theoretically impossible to deviate from the target band. Between 15-50% power the deviations are more significant and are accumulated at 1/2 of their actual time. Above 50% the deviations are most significant and their time is accumulated on a one for one time basis.

Strict control of the flux difference (and rod position) is not as necessary during part power operation because xenon distribution control at part power is less significant than control at full power. Allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during PHYSICS TESTS or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The consequences of being outside the ±5% target band but within the Figure TS.3.10-6 limit for power levels between 50% and 90% has been evaluated and determined to result in acceptable $F_Q(Z)$ values. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference. In all cases the ±5 percent target band is the LIMITING CONDITION FOR OPERATION. Only when the target band is violated do the limits under Figure TS.3.10-6 apply.

If, for any reason, the indicated axial flux difference is not controlled within the ±5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at or below 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

C. Quadrant Power Tilt Limits

QUADRANT POWER TILT limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

C. Quadrant Power Tilt Limits (continued)

core limits protected per Specification 3.10.E. A quadrant tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are set to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod.

Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During start-up and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present.

The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The percentage QUADRANT POWER TILT of 2% at which remedial and corrective action is required has been set so as to provide DNB and linear heat generation rate protection with x-y power tilts. Analyses have shown that percentage increases in the x-y power peaking factor are less than or equal to twice the increase in the indicated quadrant power tilt.

An increase in F_Q^N is not likely to occur with tilts up to 3% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q^N occurs.

Therefore, a limiting power tilt of 3 percent can be tolerated. However, a measurement uncertainty is associated with the indicated quadrant power tilt. Thus, allowing for a low measurement of power tilt, the action level of indicated tilt has been set at 2 percent. An alarm is set to alert the operator to an indicated tilt of 2 percent or greater for which action is required. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the actual tilt with in-core mappings or to determine and correct the cause of the tilt.

Should this action not be taken, the margin for uncertainty in F_Q^N is reinstated by reducing the power by 2 percent for each percent of tilt above 1.0, in accordance with the relationship described above, or as required by the restriction on peaking factors.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

C. Quadrant Power Tilt Limits (continued)

The upper limit on the quadrant tilt at which hot shutdown is required has been set so as to provide protection against excessive linear heat generation rate. The ratio of overpower to normal operation is approximately 1.15. Since the x-y component of F_Q^N is bounded by the above described relation with indicated quadrant tilt, the overpower linear heat generation rate can be avoided if the indicated tilt is restricted below 7 percent.

D. Rod Insertion Limits

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. The available control rod reactivity (or excess beyond needs) decreases with decreasing boron concentration. The negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low since the power defect increases with core burnup.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10 D.) is to measure the worth of all rods less the worth of the the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

An evaluation has been made of anticipated transients and postulated accidents, assuming that they occur during the portion of this test when the reactor is critical with all but one full-length control rod fully inserted. Further, the withdrawn full-length rod is assumed not to trip. As a result of this evaluation, it has been determined that for a steam line break upstream of the flow restrictor, the possibility of core DNB exists. However, even if core damage does result, any core fission product release would be low because of the low fission product inventory during initial startup physics testing; and further, would be contained within the reactor coolant system.

Thus, for the initial startup PHYSICS TESTS, this test will not endanger the health and safety of the public even in the event of highly improbable accidents coupled with the failure of the withdrawn control rod to trip. To perform this test later in life is equally valuable,

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

D. Rod Insertion Limits (continued)

as stated above. Therefore, this specification has been written to further minimize the likelihood of any hypothesized event during the performance of these tests later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

E. Rod Misalignment Limitation

Rod misalignment requirements are specified to ensure that power distributions more severe than those assumed in the safety analyses do not occur.

F. Inoperable Rod Position Indicator Channels

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. A misalignment less than 15 inches does not lead to over-limit power peaking factors. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or core thermocouples, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 15-inch misalignment would have no effect on power distributions. Therefore, it is necessary to apply the indirect checks following significant rod motion.

G. Inoperable Rod Limitations

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The four-week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

H. Rod Drop Time

The required drop time to dashpot entry is consistent with the safety analysis.

I. Monitor Inoperability Requirements

If either the rod bank insertion limit monitor or rod position deviation monitor are inoperable, additional surveillance is required to ensure adequate shutdown margin is maintained.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to QUADRANT POWER TILT must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

J. DNB Parameters

The RCS flow rate, T_{avg}, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the specification value indicated, shutdown is required to investigate adequacy of core cooling during operation.

For fuel regions with high burnups, the depletion of fissile nuclides and buildup of fission products greatly reduces power production capability. These combined burnup effects reduce $F^N_{\Delta H}$ sufficiently to cover residual rod bow penalties beyond a region average burnup of 40,000 MWD/MTU.

3.11 CORE SURVEILLANCE INSTRUMENTATION

Bases

The moveable detector system is used to measure the core fission power density distribution. A power map made with this system following each fuel loading will confirm the proper fuel arrangement within the core. The moveable detector system is designed with substantial redundancy so that part of the system could be out of service without reducing the value of a power map. If the system is severely degraded, large measurement uncertainty factors must be applied. The uncertainty factors would necessarily depend on the operable configuration.

Two detector thimbles per quadrant are sufficient to provide data for the normalization of the excore detector system's axial power offset feature.

The core thermocouples provide an independent means of measuring the balance of power among the core quadrants. If one excore power channel is out of service, it is prudent to have available an independent means of determining the quadrant power balance.

The moveable detector system and the thermocouple system are not integral parts of the reactor protection system. These systems are, rather, surveillance systems which may be required in the event of an abnormal occurrence such as a power tilt or a control rod misalignment. Since such occurrences cannot be predicted a priori, it is prudent to have the surveillance systems in an OPERABLE state.

3.12 SNUBBERS

Bases

All snubbers are required to be OPERABLE above COLD SHUTDOWN to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

3.13 CONTROL ROOM AIR TREATMENT SYSTEM

Bases

The Control Room Special Ventilation System is designed to filter the Control Room atmosphere during accident conditions. The system is designed to automatically start on a high radiation signal in the ventilation air or when a Safety Injection signal is received from either unit. Two completely redundant trains are provided.

Each train has a filter unit consisting of a prefilter, HEPA filters, and charcoal adsorbers. The HEPA filters remove particulates from the Control Room atmosphere and prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to remove any radioiodines from the Control Room atmosphere. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at lease 90% under test conditions more severe than expected accident conditions. System flows should be near their design values. The verification of these performance parameters combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the Control Room Special Ventilation System will perform as predicted in reducing potential doses to plant personnel below those levels stated in Criterion 19 of Appendix A to 10 CFR 50.

In-place testing procedures will be established utilizing applicable section of ANSI N510 - 1975 standard as a procedural guideline only.

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect the control room personnel and is consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release", February, 1975.

The Control Room Special Ventilation System remains OPERABLE if the ventilation system can be operated in the recirculation mode.

3.14 FIRE DETECTION AND PROTECTION SYSTEMS

Bases

Ionization, photoelectric, and thermal type fire detectors are located throughout safety related structures. These detectors sense the products of combustion during the very early stages of a fire or the heat emitted by a fire. The detectors in each area initiate an alarm in the control room. The specifications require a minimum number of detectors to be OPERABLE in each area. If this number is not OPERABLE, except for fire detectors located in primary containment, a patrolling fire watch is established in the affected area.

If an area is found to have an inoperable detector, the alarm for the affected zone may be bypassed while the detector is being repaired. Primary containment detectors are unique since (1) they are inaccessible during normal operation, and (2) no significant fire hazard exists inside containment during normal operation. Inoperable fire detectors located inside containment will be repaired during the first scheduled outage following discovery. Safety related fire detection instruments are listed in Table TS.3.14.1.

The FIRE SUPPRESSION WATER SYSTEM is supplied from the Mississippi River by two horizontal centrifugal fire pumps rated at 2000 gpm at 120 psig. One pump is motor driven and the other pump is diesel driven. A third pump also rated at 2000 gpm at 120 psig, is assigned to the screen wash system, and serves as a backup to the fire suppression water system. Header is maintained between 108 and 113 psig by a jockey fire pump. If the water demand is such that the jockey pump cannot maintain the header pressure, the screen wash pump will start (if not running) and the screen wash to fire header bypass valve will open at 102 psig. The bypass line is orificed to restrict flow to 450 gpm. On further demand, the motor driven fire pump will automatically start at 95 psig. If further demand of water is called for and the header pressure drops to 90 psig, the diesel driven fire pump will start. Pumps are designed to pump 2000 gpm and maintain a minimum of 65 psig in the fire header, measured at the highest point in the system. The screen wash pump may be directly aligned to the fire header by manual action from the control room. Any one fire pump, or the screen wash pump, can be used to supply all fire fighting water requirements. In the event that a pump is inoperable, up to seven days are allowed to restore the pump to operability or a report must be submitted to the Commission explaining the circumstances. If all pumps are inoperable, or if the fire suppression water system is incapable of supplying water to a safety related area, a backup FIRE SUPPRESSION WATER SYSTEM must be established within 24 hours and the Commission must be informed.

The cooling water system, also supplied by the Mississippi River, provides additional redundancy to the FIRE SUPPRESSION WATER SYSTEM. Crossover water supplies from the cooling water system to the fire protection system are provided for the safety related areas.

3.14 FIRE DETECTION AND PROTECTION SYSTEMS

Bases continued

Water deluge or wet pipe sprinkler systems are provided in safety related areas where a significant fire hazard exists, except for the relay and cable spreading room. Due to the nature of the equipment in the relay and cable spreading area, a carbon dioxide system is provided. Whenever a deluge or sprinkler system is inoperable, a continuous fire watch with backup fire suppression equipment available is stationed in the area until OPERABILITY is restored. Whenever the relay and cable spreading room carbon dioxide systems becomes inoperable, up to 14 days are allowed to complete maintenance. If the system cannot be restored to OPERABLE status within this time period, a report outlining the situation is submitted to the Commission. Whenever the carbon dioxide system is inoperable, a continous fire watch with backup fire suppression equipment is stationed in the room. Since the relay and cable spreading area is occupied during normal working hours, the automatic initiation feature of the CO_2 system is bypassed during this period and whenever entry is made during other times. The system is initiated manually in the event fire is detected when the room is occupied.

In addition to deluge and sprinkler systems, hydrant hose houses are located in the yard and hose stations are located throughout the plant. These hose stations provided primary and backup protection for safety related systems and components. Normally all yard hydrant hose houses and hose stations are OPERABLE when a reactor is above COLD SHUTDOWN. If a hose house or station protecting safety related equipment becomes inoperable, additional hose must be available for routing to the unprotected area. This hose may be supplied from an OPERABLE hydrant hose house, hose station, or brigade locker.

Piping and electrical penetrations are provided with seals where required by the fire severity. If a seal is made or found to be inoperable for any reason, the penetration area is continuously attended or monitored hourly if fire detectors on at least one side of the inoperable barrier are OPERABLE until an effective fire seal is restored. Seals have been qualified for the maximum fire severity present on either side of the barrier.

3.15 EVENT MONITORING INSTRUMENTATION

Bases

The OPERABILITY of the event monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578 and NUREG-0737.

BASES FOR

SECTION 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

4.1 OPERATIONAL SAFETY REVIEW

Bases

Channel Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequencies set forth are deemed adequate for reactor and steam system instrumentation.

Channel Calibration

Calibration is performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels daily calibration against a thermal power calculation will account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" efforts induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Channel Functional Tests

Minimum testing frequency is based on evaluation of unsafe failure rate data and reliability analysis. This is based on operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bonafide signal. The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hr per channel.

4.1 OPERATIONAL SAFETY REVIEW

Bases continued

Channel Functional Tests (continued)

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

$$A = W - Q \left(\frac{W}{N-M+2}\right) = 1 - \frac{N!(\lambda W)^{N-M+1}}{(N-M+2)!(M-1)!}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999968, assuming a channel failure rate, λ, equal to 2.5×10^{-6} hr^{-1} and a test interval, W, equal to 720 hours.

This average availability of the 2-out-of-3 system is high, hence the test interval of one month is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

Channel Response Tests

Measurement of response times for protection channels are performed to assure response times within those assumed for accident analysis (USAR, Section 14).

4.2 INSERVICE INSPECTION AND TESTING OF PUMPS AND VALVES REQUIREMENTS

Bases

The inservice inspection program for the Prairie Island plant conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical the inspection of ASME Code Class 1, Class 2, and Class 3 components is performed in accordance with Section XI of the ASME Code. If a code required inspection is impractical for the Prairie Island facility, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10 year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

4.3 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Bases

The requirement for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA which bypasses containment.

4.4 CONTAINMENT SYSTEM TESTS

Bases

The containment system consists of a steel containment vessel, a concrete shield building, the auxiliary building special ventilation zone (ABSVZ), a shield building ventilation system, and an auxiliary building special ventilation system. In the event of a loss-of-coolant accident, a vacuum in the shield building annulus will cause most leakage from the containment vessel to be mixed in the annulus volume and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum. Some of the leakage goes to the ABSVZ from which it is exhausted through a filter. A small fraction bypasses both filter systems.

The freestanding containment vessel is designed to accommodate the maximum internal pressure that would result form the Design Basis Accident (Reference 1). For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safeguards results in a peak pressure of less than 46 psig at 268°F.

The containment will be strength-tested at 51.8 psig and leak-tested at 46.0 psig to meet acceptance specifications.

The safety analysis (References 2, 3) is based on a conservatively chosen reference set of assumptions regarding the sequence of events relating to activity release and attainment and maintenance of vacuum in the shield building annulus and the auxiliary building special ventilation zone, the effectiveness of filtering, and the leak rate of the containment vessel as a function of time. The effects of variation in these assumptions, including that for leak rate, has been investigated thoroughly. A summary of the items of conservatism involved in the reference calculation and the magnitude of their effect upon off-site dose demonstrates the collective effectiveness of conservatism in these assumptions.

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 vent 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 vent 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.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

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 inleakage (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, achieved either by normal system operation or hydrostatically testing 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.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

The shield building ventilation system consists of two independent systems that have only a discharge point in common, the shield building vent. Both systems are normally activated and one alone must be capable of accomplishing the design function of the system. During the first operating cycle, tests were performed to demonstrate the capability of the separate and combined systems under different wind conditions. During quarterly operability tests, the drawdown transient of shield building pressure is compared to the computed predicted drawdown transient for non-accident conditions and leakage equal to 75% of Figure TS.4.4-1 (840 cfm at -2.0 INWG). The -2.0 INWG setpoint of the recirculation damper must be reached and the equilibrium pressure in the annulus must be less than -1.82 INWG to demonstrate adequate shield building leak tightness.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at lease once per operating cycle to verify operability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. A charcoal adsorber tray which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration of the system adsorbent through its life is currently under development. When this tray is available, sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for the methyl iodide removal efficiency tests. Each module withdrawn will be replaced or blocked off. Until these trays can be installed, to guarantee a representative adsorbent sample, procedures should allow for the removal of a tray containing the oldest batch of adsorbent in each train, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. One sample will be submitted for laboratory analysis and the other held as a backup. If test results are unacceptable, all adsorbent in the train will be replaced. Adsorbent in the tray removed for sampling will be renewed. Any HEPA filters found defective will be replaced. Replacement charcoal adsorber and HEPA filters will be qualified in accordance with the intent of Regulatory Guide 1.52 - Rev. 1 June 1976.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

If significant painting, fire, or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals, or foreign material, the same tests and sample analysis will be performed as required for operational use.

Operation of each train of the system for 10 hours every month will demonstrate OPERABILITY of the system and remove excessive moisture which may build up on the adsorber.

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A minimum containment shell temperature of 30°F has been specified to provide assurance that an adequate margin above NDTT exists. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in COLD SHUTDOWN. Requiring containment shell temperature to be verified to be above 30°F prior to plant heatup from COLD SHUTDOWN provides assurance that this temperature is above NDTT prior to establishing conditions requiring CONTAINMENT INTEGRITY (Reference 7).

A maximum temperature differential between the average containment and annulus air temperatures of 44°F has been specified to provide assurance that offsite doses in the event of an accident remain below those calculated in the FSAR. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in COLD SHUTDOWN. Requiring this temperature differential to be verified to be less than 44°F prior to plant heatup from COLD SHUTDOWN provides assurance that this parameter is within acceptable limits prior to establishing conditions requiring CONTAINMENT INTEGRITY (Reference 7).

References

1. USAR, Section 5 and FSAR, Appendix 14-C
2. USAR, Section 14 and FSAR, Appendix G
3. Safety Evaluation Report, Sections 6.2 and 15.0
4. USAR, Section 14
5. USAR, Section 5.4.3
6. Letter to NSP from AEC dated November 29, 1973
7. NSP Report, "Prairie Island Containment Systems Special Analyses", dated April 9, 1976.

4.5 ENGINEERED SAFETY FEATURES

Bases

The Safety Injection System and the Containment Spray System are principal plant Safety Systems that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring OPERABILITY of these systems is therefore to combine systems tests to be performed during refueling shutdowns, with more frequent component tests which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry (Reference 1).

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked weekly, and the initiating circuits are tested monthly (in accordance with Specification 4.1).

The program of pump and valve testing for safety related equipment conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, tests of ASME Code Class 1, Class 2, and Class 3 pumps and valves are performed in accordance with Section XI of the ASME Code. If a code required test is impractical for the Prairie Island facility, a request for deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 20-month inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the 20-month inspection period. Negative reports will not be made.

Other systems that are also important to the emergency cooling function are the accumulators, the component cooling system, the Cooling Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

4.5 ENGINEERED SAFETY FEATURES

Bases continued

The purpose of the surveillance requirements on ECCS throttle valves is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point in the High Head Safety Injection System is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Reference

1. USAR, Section 6.2.1

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEMS

Bases

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and to carry load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be alarm-indicated without need for test startup.

The less frequent overall system test will demonstrate that the emergency power system and the control systems for the engineered safeguards equipment will function automatically in the event of loss of all other sources of a-c power, and that the diesel generators will start automatically in the event of a loss-of-coolant accident. This test will demonstrate proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment, as well as the OPERABILITY of the diesel generators.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency will be detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function. Its possible failure to respond is, of course, anticipated by providing two diesel generators, each supplying, through an independent bus, a complete and adequate set of engineered safeguards equipment. Further, both diesel generators are provided as backup to multiple sources of external power, and this multiplicity of sources should be considered with regard to adequacy of test frequency.

Each diesel generator can start and be ready to accept full load within 10 seconds, and will sequentially start and supply the power requirements for one complete set of safeguards equipment in approximately one minute (Reference 1).

An internal fault in the generator could damage the generator severely. Moreover, this change complies with BTP EICSB 17. Auto-connected loads should not exceed the overload rating of the diesel generator for the 2000 hour maintenance interval, as prescribed in Regulatory Guide 1.9.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide indication of a cell becoming unserviceable long before it fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEMS

Bases continued

The surveillance specified for the pressurizer heater power source provides assurance that Backup Heater Group "B" can be transferred to its emergency bus. Normally, this group of heaters is supplied from a normal plant 480 volt bus. In an emergency, a manual transfer switch can be used to supply the heater group from a safeguards supply bus.

Reference

1. USAR, Section 8.4

4.7 MAIN STEAM ISOLATION VALVES

Bases

The main steam isolation valves serve to limit the cooldown rate of the reactor coolant system and the reactivity insertion that could result from a main steam break incident (Reference 1). Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds is selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis (Reference 2).

References

1. USAR, Section 14.5.5
2. USAR, Section 14.5.5

4.8 STEAM AND POWER CONVERSION SYSTEMS

Bases

Monthly testing of the auxiliary feedwater pumps, monthly valve inspections, and startup flow verification provide assurance that the AFW system will meet emergency demand requirements. The discharge valves of the pumps are normally open, as are the suction valves from the condensate storage tanks. Proper opening of the steam admission valve on each turbine-driven pump will be demonstrated each time a turbine-driven pump is tested. Ventilation system isolation dampers required to function for the postulated rupture of a high energy line will also be tested.

At 18-month intervals, pump starting and valve positioning is verified using test signals to simulate each of the automatic actuation parameters.

Reference

USAR, Sections 11.9, 14, and FSAR, Appendix I.

4.9 REACTIVITY ANOMALIES

Bases

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As POWER OPERATION proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

Reference

USAR, Section 3.3.2

4.10 RADIATION ENVIRONMENTAL MONITORING PROGRAM

Bases

A. Sample Collection & Analysis

The Radiation Environmental Monitoring Program required by this specification provides measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the plant operation. This program thereby supplements the radiological effluent monitoring by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the bases of the effluent measurements and modeling of the environmental exposure pathways.

The detection capabilities required by Table 4.10-2 are state-of-the-art for routine environmental measurements in industrial laboratories, and the LLDs for drinking water meet the requirement of 40 CFR 141.

B. Land Use Census

This specification is provided to ensure that changes in the use of off site areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), 2) a vegetation yield of 2 kg/square meter.

C. Interlaboratory Comparison Program

The requirement for participation in an interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as a part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

4.11 RADOIACTIVE SOURCE LEAKAGE TEST

Bases

Licensee's program, facilities, personnel. and procedures for safe storage, handling, and use of sealed sources containing radioactive materials is described in USAR Section 12.4. The surveillance program described in this specification is a part of licensee's program to detect and control contamination of areas in the plant by such radioactive materials. Small quantities of byproduct materials are exempt for licensing by 10 CFR 30.18 and therefore are exempt from leakage tests in this specification. Inhalation or ingestion of such small quantities of byproduct materials from a sealed source would result in less than one maximum permissible body burden for total body irradiation. Sources containing less than 0.1 microcurie of plutonium are exempt from leakage tests by 10 CFR 70.39(c) and therefore such quantities of special nuclear materials (including alpha emitters) are exempt from leakage tests in this specification. The acceptance criteria of less than 0.005 microcuries on the test sample is also based on 10 CFR 70.39(c).

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameters found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameters, localized corrosion would most likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1.0 gpm). Cracks having a primary-to-secondary leakage less than 1.0 gpm during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents (Reference 1). Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected by radation monitors of steam generator blowdown. Leakage in excess of 1.0 gpm will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged or sleeved.

Wastage-type defects are unlikely with proper chemistry treatment of secondary coolant. However, even if this type of defect occurs it will be found during scheduled in-service steam generator tube inspections. Repair or plugging will be required of all tubes with imperfections that could develop defects having less than the minimum acceptable wall thickness prior to the next inservice inspection which, by the definition of Specification 4.12.D.1.(f), is 50% of the tube or sleeve nominal wall thickness. Wastage type defects having a wall thickness greater than 0.025 inches will have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents (Reference 1). Steam generator tube inspections of operating

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050-inch wall thickness (Reference 2).

Whenever the results of any steam generator tubing in-service inspection fall into Catgegory C-3, these results will be promptly reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

Degraded steam generator tubes may be repaired by the installation of sleeves which span the section of degraded steam generator tubing. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded.

The following sleeve designs have been found acceptable by the NRC Staff:

a. Westinghouse Mechanical Sleeves (WCAP 10757)

b. Westinghouse Brazed Sleeves (WCAP-10820)

c. Combustion Engineering Leak Tight Sleeves (CEN-294-NP)

Descriptions of other future sleeve designs shall be submitted to the NRC for review and approval prior to their use in the repair of degraded steam generator tubes. The submittals related to other sleeve designs shall be made at least 90 days prior to use.

References

1. Testimony of J Knight in the Prairie Island Public Hearing on 1/28/75

2. Testimony of L Frank in the Prairie Island Public Hearing on 1/28/75

4.13 SNUBBERS

Bases

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of 10% of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

All safety-related snubbers installed or planned for use at Prairie Island are hydraulic snubbers. No mechanical snubbers are used.

4.14 CONTROL ROOM AIR TREATMENT SYSTEM TESTS

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. A charcoal adsorber tray which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration of the system adsorbent through its life is currently under development. When this tray is available, sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for methyl iodide removal efficiency tests. Each module withdrawn will be replaced or blocked off. Until these trays can be installed, to guarantee a representative adsorbent sample, procedures should allow for the removal of a tray containing the oldest batch of adsorbent in each train, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. One sample will be submitted for laboratory analysis and the other held as a backup. If test results are unacceptable, all adsorbent in the train shall be replaced. Adsorbent in the tray removed for sampling shall be renewed. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 - Rev. 1 June 1976.

If significant painting, fire, or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals, or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the shift supervisor after consulting knowledgeable staff members.

Operation of each train of the system for 15 minutes every month will demonstrate OPERABILITY of the system and remove excessive moisture which may build up on the adsorber.

Demonstrating automatic initiation of the system using simulated accident signals will assure that the system will start when required.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

4.15 SPENT FUEL POOL SPECIAL VENTILATION SYSTEM

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. A charcoal adsorber tray which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration of the system adsorbent through its life is currently under development. When this tray is available, sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for methyl iodide removal efficiency tests. Each module withdrawn will be replaced or blocked off. Until these trays can be installed, to guarantee a representative adsorbent sample, procedures should allow for the removal of a tray containing the oldest batch of adsorbent in each train, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. One sample will be submitted for laboratory analysis and the other held as a backup. If test results are unacceptable, all adsorbent in the train shall be replaced. Adsorbent in the tray removed for sampling shall be renewed. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 - Rev. 1 June 1976.

If significant painting, fire, or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals, or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant will be made by the shift supervisor after consulting knowledgeabl staff members.

Operation of each train of the system for 10 hours every month will demonstrate OPERABILITY of the system and remove excessive moisture which may build up on the adsorber.

Demonstrating automatic initiation of the system using simulated accident signals will assure that the system will start when required.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

4.16 FIRE DETECTION AND PROTECTION SYSTEMS

Bases

The minimum number of fire detectors required to be OPERABLE in each fire zone are functionally tested following the manufacturer's recommendations each six months, except for those located inside the primary containment which are tested during each COLD SHUTDOWN exceeding 24 hours unless performed during the previous six months. These tests tests are performed by the plant staff. Other fire detectors will be tested at an interval which experience has shown to be necessary to assure reliable operation. Every six months an alarm circuit check is performed. This check can be performed in conjunction with detector functional tests. All circuitry is also provided with automatic supervision for opens and ground faults.

Fire pumps are tested each month to verify OPERABILITY. Test starting of the screen wash pump is not required since it is normally in service. Each fire pump is manually started and operated for at least 15 or 20 minutes with pump flow directed through the recirculation test line. Every 18 months the OPERABILITY of the automatic actuation logic for the fire pumps and the screen wash pump is verified and the performance of each pump is verified to meet system requirements. The specified flush and valve lineup check provide assurance that the piping system is capable of supplying fire suppression water to all safety related areas. When one of the pumps is inoperable the operable pumps are run daily to verify OPERABILITY until all pumps are once again available.

FIRE SUPPRESSION WATER SYSTEM flow tests will be done at least every three years to verify hydraulic performance. The testing will be performed using Section II, Chapter 5 of the Fire Protection Handbook, 14th Edition, as a procedural guide. The test is generally performed in conjunction with insurance inspections.

Surveillance specified for each spray and sprinkler system is intended to assure that the systems will function as designed when they are needed. Functional tests are conducted at 18 month intervals on those systems provided with test facilities.

The testing specified for the relay and cable spreading room CO_2 system provides assurance that the CO_2 inventory is adequate to extinguish a fire in this area and that the system is capable of automatic actuation.

Hose stations and yard hydrant hose houses are inspected monthly to verify that all required equipment is in place. Gaskets in hose couplings are inspected periodically and the hose is pressure tested. Pressure testing of outdoor hose is conducted more frequently than indoor hose because of the less favorable storage conditions. OPERABILITY of hose station isolation valves is verified every three years by partially opening each valve to verify flow. All of these tests provide a high degree of assurance that each hose station will perform satisfactorily after periods of standby service.

4.16 FIRE DETECTION AND PROTECTION SYSTEMS

Bases continued

Plant fire barrier walls are provided with seals for pipes and cables where necessary. Where such seals are installed, they must be maintained intact to perform their function. Visual inspection of each installed seal is required every 18 months and after seal repair. A visual inspection following repair of a seal in the secondary containment boundary is sufficient to assure that seal leakage will be within acceptable limits.

4.17 RADIOACTIVE EFFLUENTS SURVEILLANCE

Bases

Radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments are calculated in accordance with the methods in the ODCM to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY requirements and use of this instrumentation are consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY requirements for this instrumentation are consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The dose calculations for liquid effluents in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", Revision 1, April 1977. NUREG-0133, October, 1978, provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

4.17 RADIOACTIVE EFFLUENTS SURVEILLANCE

Bases continued

The dose calculations for gaseous effluents in the ODCM also implement the requirements of Section III.A that conformance with the guides be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977. The ODCM equations provided for determining the air doses at the SITE BOUNDARY will be based upon the historical average atmospheric conditions. NUREG-0133, October, 1978 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

4.18 REACTOR COOLANT VENT SYSTEM PATHS

Bases

The manual valves in the reactor coolant vent system are blocked and tagged in the open position to eliminate the possibility that operation of the vent system could be blocked by the inadvertent closure of any of the vent system manual valves.

The cycling of each solenoid operated vent valve once each refueling ensures the ability of these valves to open if required to vent the reactor coolant system. More frequent cycling of the valves is not practical because they cannot be isolated from the reactor coolant system while the plant is operating.

Reactor coolant vent system flow will be determined qualitatively as part of the normal fill and vent procedures following each refueling. This will assure that there are no blockages in the reactor coolant vent system piping.

EXHIBIT C

SAFETY EVALUATION OF 7.0% vs 6.9% NEGATIVE FLUX RATE TRIP SETPOINT

The negative flux rate trip is designed to protect against fuel damage resulting from dropped RCCA events. NSP has previously submitted dropped rod transient analysis methods based on a setpoint of 6.9% with a time constant of 2.0 seconds. This evaluation describes the effect of changing the method to include the 7.0% setpoint.

The method for analyzing dropped rod events is divided into two separate parts. The first part is to determine which rods will cause a reactor trip on negative flux rate. Whether or not a particular rod causes a trip is dependent on the rod worth, relative core tilts, reactor feedback characteristics and the trip setpoint.

For example, at a 6.9% setpoint, Prairie Island 1 Cycle 10 will trip at rod worths greater than approximately 120 pcm. Raising the setpoint to 7.0% increases the rod worth required to trip the reactor to about 125 pcm. This first part of the dropped rod analyses has been performed for the current operating cycles at Prairie Island (Unit 1 Cycle 10 and Unit 2 Cycle 10) at a setpoint of 7.0% and will be analyzed at the 7.0% value for all future cycles.

The second step in the dropped rod analyses is to determine the consequences of dropped rods which do not cause a reactor trip. Dropped rods induce core tilts which can cause the automatic rod controller to "overshoot" the initial power and end up at a power greater than 100%. This overshoot of power, combined with the skewed power distribution, can lead to the reactor core approaching MDNBR safety limits. NSP has performed this second part of the analyses on what is expected to be a bounding basis. Dropped rod worths and reactor conditions much more severe than expected have been used, and all transient acceptance criterion are met. Results of this analysis are shown in Table 1 and Figure 1-5. The rod worth that do not cause a trip at the 6.9% setpoint and at the 7.0% setpoint have been verified to be bounded by the current anlaysis. If it is found in future cycles that dropped rods cause conditions to be outside those assumed in the current analysis, the analysis will be repeated using the new conditions to ensure all acceptance criterion are still met.

TABLE 1

PRAIRIE ISLAND NUCLEAR PLANT DROPPED ROD TRANSIENT RESULTS

(Calculated Value / Acceptance Criteria)

MDNBR	> 1.4 / 1.3
Peak RCS Pressure (psia)	2234 / 2750
Peak MSL Pressure (psia)	756 / 1210

Prairie Island Nuclear Plant
Dropped Rod - Automatic Control

Figure 1
K Effective

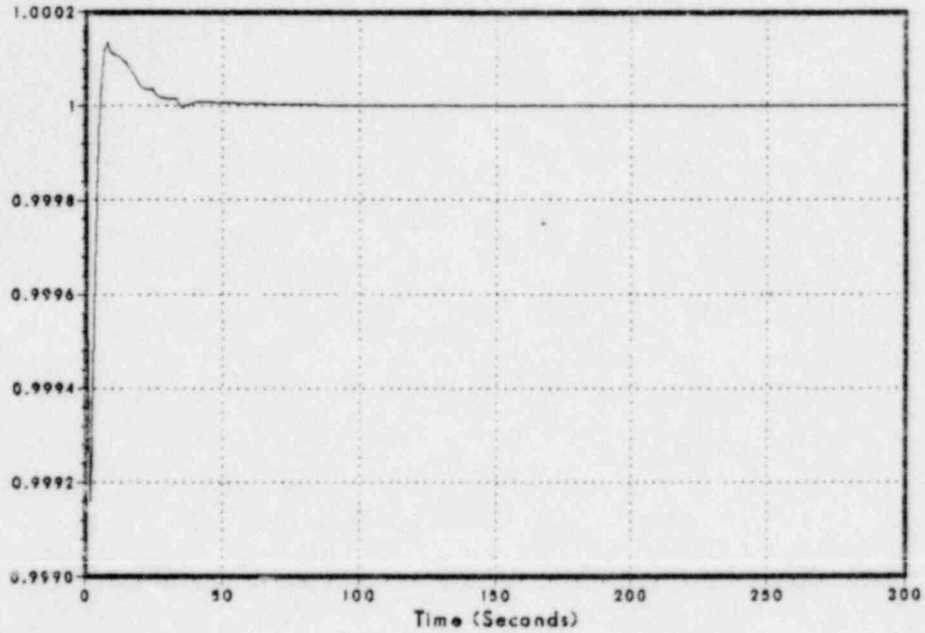
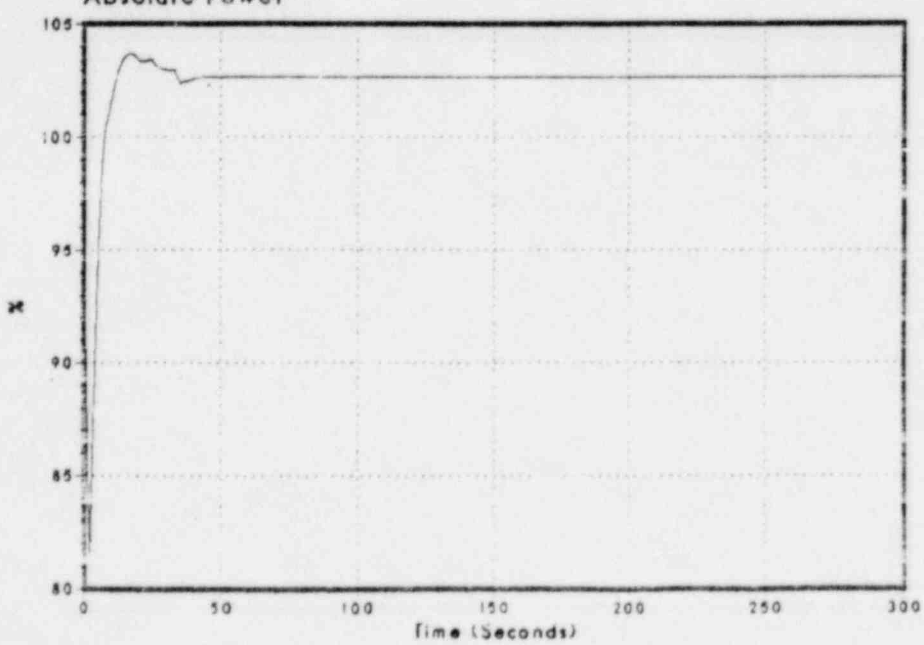


Figure 2
Absolute Power



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Prairie Island Nuclear Plant
Dropped Rod - Automatic Control

Figure 3
Core Average Heat Flux

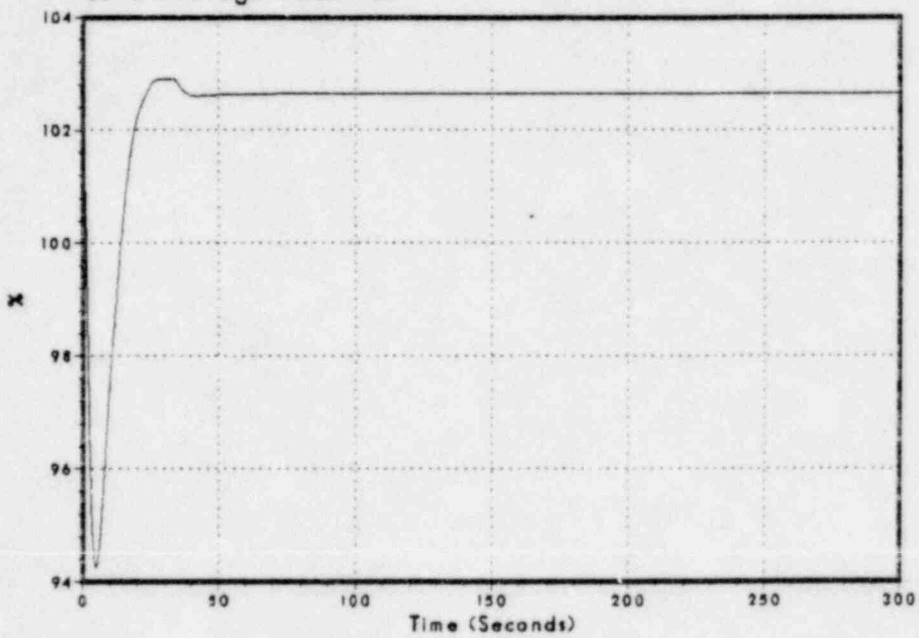
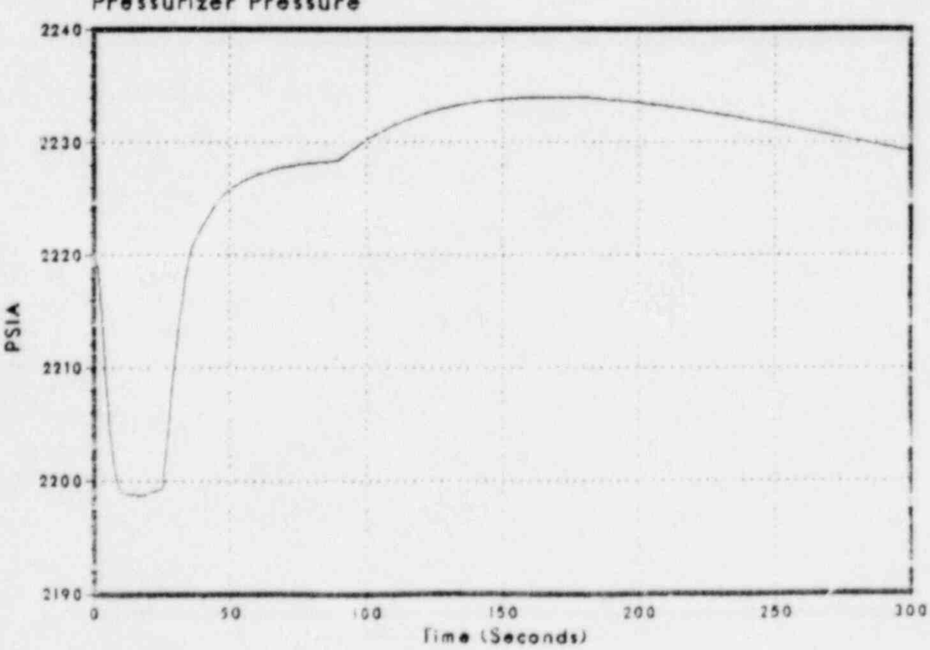


Figure 4
Pressurizer Pressure



NSP

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Prairie Island Nuclear Plant
Dropped Rod – Automatic Control

Figure 5
Vessel Average Temperature

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