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

October 27, 1989

Posted
Amdt. 84 to DPR-60
(1) (See Correction letter of 11-9-89)
(2) (See Correction letter of 3-7-90)

Dockets Nos. 50-282
and 50-306

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENTS NOS. 91 AND 84 TO FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60: TECHNICAL SPECIFICATION (TS) UPGRADE (TAC NOS. 61081 AND 61082)

The Commission has issued the enclosed Amendments Nos. 91 and 84 to Facility Operating Licenses No. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TS) in response to your application dated March 17, 1986, and supplemented by letter dated July 27, 1987.

The amendments change the TS by revising TS sections 1, 2, 3, 4, 5 and 6 in support of the human error reduction program. The objective of the changes associated with these amendments is to reduce the potential for human error resulting from lack of complete guidance associated with the requirement of the TS. In order to achieve this objective, changes include the reorganization and standardization of some TS sections, the addition of action statements for limiting conditions for operation (LCO) and removal of ambiguities. Because the issuance of the amendments results in voluminous changes to the TS, time will be needed to train operating personnel and revise plant operating procedure. Therefore, the amendments will become effective 105 days from the date of issuance for both units.

Copies of the Safety Evaluation related to the amendments and Notice of Issuance are also enclosed.

The issuance of these amendments completes our work effort under TACs Nos. 61081 and 61082.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 91 to
   License No. DPR-42
2. Amendment No. 84 to
   License No. DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

The issuance of these amendments completes our work effort under TACs Nos. 61081 and 61082.

Sincerely,

ORIGINAL SIGNED BY D. DiIANNI

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

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License No. DPR-60
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cc w/enclosures:
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Mr. T. M. Parker
Northern States Power Company

Prairie Island Nuclear Generating Plant

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. DPR-42

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

   A. The applications for amendment by Northern States Power Company (the licensee) dated March 17, 1986 and supplemented by letter dated July 27, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.91 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 105 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John O. Thoma

John O. Thoma, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 27, 1989

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.84
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company (the licensee) dated March 17, 1986 and supplemented by letter dated July 27, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 84 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 105 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John O. Thoma

John O. Thoma, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 27, 1989

ERRATA

ATTACHMENT TO LICENSE AMENDMENTS NOS. 91 AND 84
FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR 60
DOCKETS NOS. 50-282 AND 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
TS-i thru x	TS-i thru xiii
TS.1-1 thru 8	TS.1-1 thru 8
TS.2.1-1	TS.2.1-1
TS.2.1-2	-----------
TS.2.2-1	TS.2.2-1
TS.2.2-2	-----------
TS.2.3-1 thru 4	TS.2.3-1 thru 4
TS.2.3-5 thru 6	-----------
-----------	TS.3.0-1
TS.3.1-1 thru 21	TS.3.1-1 thru 12
Move Figures TS.3.1-1,2 to between TS.3.1-7 and 8	
Move Figures TS.3.1-3 to between TS.3.1 thru 10 and 11	
TS.3.2-1 thru 5	TS.3.2-1 thru 2
TS.3.3-1 thru 8	TS.3.3-1 thru 8
TS.3.4-1 thru 3	TS.3.4-1 thru 3
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TABLE TS.3.5-4	TABLE TS.3.5-4
TS.3.6-1 thru 6	TS.3.6-1 thru 4
TS.3.7-1 thru 3	TS.3.7-1 thru 3
TS.3.8-1 thru 6	TS.3.8-1 thru 4
TS.3.9-1 thru 13	TS.3.9-1 thru 7
TS.3.10-1 thru 17	TS.3.10-1 thru 8
Figure TS.3.10-1	Figure TS.3.10-1
TS.3.11-1 thru 2	TS.3.11-1
TS.3.12-1	TS.3.12-1
TS.3.13-1 thru 2	TS.3.13-1 thru 2
TS.3.14-1 thru 6	TS.3.14-1 thru 4
TS.3.15-1 thru 2	TS.3.15-1 thru 2
TS.4.1-1 thru 3	TS.4.1-1
TABLE TS.4.1-2B(page 2 of 2)	TABLE TS.4.1-2B(page 2 of 2)
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TS.4.3-2	--------
TS.4.4-3	TS.4.4-3
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TS.4.6-1 thru 3	TS.4.6-1 thru 3
TS.4.7-1	TS.4.7-1
TS.4.8-1 thru 2	TS.4.8-1 thru 2
TS.4.9-1	TS.4.9-1
TS.4.10-3	---------
TS.4.11-2	TS.4.11-2
TS.4.12-6 thru 7	---------
TS.4.13-3 thru 4	TS.4.13-3
TS.4.14-1 thru 2	TS.4.14-1
TS.4.15-1 thru 2	TS.4.15-1
TS.4.16-2	TS.4.16-2

Correction letter of 3-7-90

REMOVE	INSERT
TS.4.16-5 thru 6	---------
TS.4.17-4 thru 5	TS.4.17-4
TS.4.18-1	TS.4.18-1
TS.5.1-2	TS.5.1-2
TS.6.7-1 thru 5	TS.6.7-1 thru 5

Section 2.0

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-----	B.2.2-1
-----	B.2.3-1 thru 3

Section 3.0

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Correction letter of 3-7-90

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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY

AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY shall exist 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.

4. The Auxiliary Building Special Ventilation System is 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

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

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 shall exist when:

1. Penetrations required to be isolated during accident conditions are either:

   a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specifications 3.6.C and 3.6.D.

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

3. The equipment hatch is closed and sealed.

4. Each air lock is in compliance with the requirements of Specification 3.6.M.

5. The containment leakage rates are within their required limits.

COLD SHUTDOWN

A reactor is in the COLD SHUTDOWN condition when the reactor is subcritical by at least 1% $\Delta k/k$ 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.

Correction letter of 3-7-90

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 sample) 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 shall be any 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, as specified in Section 2.3, for automatic protective devices related to those variables having significant safety functions.

MEMBERS OF THE PUBLIC

MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, 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.

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 the Radiological Environmental Monitoring Program.

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 shall be those tests performed to measure the 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 2% 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 THERMAL POWER.

PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formula, sampling, analysis, tests, 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 10 CFR Parts 20, 61 and 71 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.

PURGE-PURGING

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

QUADRANT POWER TILT RATIO

QUADRANT POWER TILT RATIO shall be 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.

RATED THERMAL POWER

RATED THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant of 1650 megawatts thermal (MWt).

REFUELING

A unit is in the REFUELING condition when:

1. There is fuel in the reactor vessel.
2. The vessel head closure bolts are less than fully tensioned or the head is removed.
3. The reactor coolant average temperature is less than or equal to 140°F, and
4. The boron concentration of the reactor coolant system and the refueling cavity is sufficient to ensure that the more restrictive of the following conditions is met:

   a. $K_{eff} \leq 0.95$, or

   b. Boron concentration ≥ 2000 ppm.

REPORTABLE EVENT

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

SHIELD BUILDING INTEGRITY

SHIELD BUILDING INTEGRITY shall exist 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.
3. The Shield Building Ventilation System is OPERABLE.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Correction letter of 3-7-90

SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

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 shall be the total reactor core heat transfer rate to the reactor coolant.

UNRESTRICTED AREAS

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which 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 for industrial, commercial, institutional and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM shall be 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 shall be 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. Vent, used in system names, does not imply a VENTING process.

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, pressurizer pressure, and the highest reactor coolant system loop average temperature shall not exceed the limits shown in Figure TS.2.1-1.

2. Whenever the point defined by the combination of the highest reactor coolant system loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line in Figure TS.2.1-1, be in at least HOT SHUTDOWN within one hour, and comply with the requirements of Specification 6.4.

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

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

2. Whenever the reactor coolant system pressure has exceeded 2735 psig with the reactor critical, be in at least HOT SHUTDOWN with the reactor coolant system pressure within its limit within one hour, and comply with the requirements of Specification 6.4.

3. Whenever the reactor coolant system pressure has exceeded 2735 psig with the reactor subcritical, reduce the reactor coolant system pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.4.

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) - $\leq 40\%$ 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 - ≤ 2385 psig.

   c. Low pressurizer pressure - ≥ 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.A.2.d Cont.

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 q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ 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

$$\Delta T_p \leq \Delta T_o \left[K_4 - \frac{K_5 t_3 s T}{1 + t_3 s} - K_6(T - T') - f(\Delta I)\right]$$

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'
t_3	=	10 sec
f(ΔI)	=	as defined in d. above

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

2.3.A.2.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$58.2 Hz

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

*The low steam generator water level in coincidence with steam/feedwater mismatch trip may be deleted following installation of the digital feedwater control system incorporating the median signal selector function. An acceptable verification and validation program shall be demonstrated to the satisfaction of the NRC Staff.

2.3.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. Low power block of single loop loss of flow is permitted whenever power range neutron flux is $\leq$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 $\leq$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 $\geq$50% of RATED THERMAL POWER.

C. Control Rod Withdrawal Stops

1. Block automatic rod withdrawal:

   a. Turbine load $\leq$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 a condition in which the equipment is not required to be OPERABLE.

If the equipment is required to be Operable above COLD SHUTDOWN, then be in:

1. At least HOT SHUTDOWN within the next 6 hours, and
2. COLD SHUTDOWN within the following 30 hours.

If the equipment is required to be OPERABLE when the reactor coolant system average temperature is above 350°F, then be in:

1. At least HOT SHUTDOWN within the next 6 hours, and
2. Reduce reactor coolant system average temperature below 350°F within the following 6 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

(1) A reactor shall not be made or maintained critical unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are in operation, except 1) during low power PHYSICS TESTS or 2) as specified in 3.1.A.1.a.(2) below.

(2) With less than the above required reactor coolant loops in operation, be in at least HOT SHUTDOWN within 6 hours.

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

(1) Reactor coolant system average temperature shall not exceed 350°F unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are OPERABLE with at least one reactor coolant loop in operation* (except as specified in 3.1.A.1.b(2) below).

(2) A reactor coolant loop may be inoperable for 72 hours provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, reduce reactor coolant system average temperature below 350°F within the next 6 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.

3.1.A.1.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, 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 prompt 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 prompt action to restore one OPERABLE method of removing decay heat.

(4) A reactor coolant pump may be started at RCS temperature less than the 310°F** 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.1.d.(2) below).

(2) With one or both residual heat removal loop(s) inoperable, prompt action shall be taken to restore the inoperable residual heat removal loop(s) to an OPERABLE status. During reduced inventory conditions, a safety injection pump may be run as required to maintain adequate core cooling and RCS inventory in the event of a loss of Residual Heat Removal System cooling.

*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.

**Valid until 20 EFPY

3.1.A.2 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 (except as specified in 3.1.A.2.a.2 and 3.1.A.2.a.3 below).

(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 6 hours.

(3) With the pressurizer otherwise inoperable, within one hour initiate the action necessary to place the unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant average temperature below 350°F within the following 6 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 15 minutes initiate the action necessary to place the 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 6 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, promptly place an OPERABLE residual heat removal loop into operation.

3.1.A.2.c Pressurizer Power Operated Relief Valves

(1) Reactor Coolant System average temperature greater than or equal to 310°F*

(a) A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 310°F* unless two power operated relief valves (PORVs) and their associated block valves are OPERABLE (except as specified in 3.1.A.2.c(1)(b) below).

(b) 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 310°F* within the following 6 hours.

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

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

(2) Reactor Coolant System average temperature below 310°F

With Reactor Coolant System temperature less than 310°F*; 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.

*Valid until 20 EFPY

3.1.A.3 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.3.b and 3.1.A.3.c below).

b. During STARTUP OPERATION and 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 any one of these conditions is not restored to an OPERABLE status within 30 days, be 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.

3.1.B. Pressure/Temperature Limits

1. Reactor Coolant System

   a. The Unit 1 and Unit 2 Reactor Coolant Systems (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 with:

      1. A maximum heatup of 60°F in any 1-hour period.

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

   b. If these conditions cannot be satisfied, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or 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.

2. Pressurizer

   a. The pressurizer temperature shall be limited to:

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

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

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

   c. If these conditions cannot be satisfied, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT SHUTDOWN within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

Correction letter of 3-7-90

3.1.B.3. Steam Generators

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

b. If these conditions cannot be satisfied, reduce the steam generator pressure to less than or equal to 200 psig within 30 minutes; perform an engineering evaluation to determine the effects of the overpressurization on the structural integrity of the steam generator; determine that the steam generator remains acceptable for continued operation prior to increasing its temperature above 200°F.

3.1.C. REACTOR COOLANT SYSTEM LEAKAGE

1. Leakage Detection

The reactor coolant system average temperature shall not exceed 200°F unless 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

The following leakage limitations are applicable whenever the reactor coolant system average temperature exceeds 200°F.

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, be in at least 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 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 initiate action to place the 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 condition is corrected.

3.1.C.2 e. If the total reactor coolant system to secondary coolant system leakage through both steam generators of a unit exceeds 1.0 gallon per minute (gpm), within one hour initiate action to place the unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours and perform an inservice steam generator tube inspection in accordance with Technical Specification 4.12.

3. <u>Pressure Isolation Valve Leakage</u>

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 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.

Correction letter of 3-7-90

3.1.D. MAXIMUM COOLANT ACTIVITY

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

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

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

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

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

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

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

4. Annual reporting requirements are identified in 6.7.A.1.c.

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

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

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

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

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

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

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

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

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

3.1.F. ISOTHERMAL TEMPERATURE COEFFICIENT (ITC)

1. When the reactor is critical, the isothermal temperature coefficient shall be less than 5 pcm/°F with all rods withdrawn, except during low power PHYSICS TESTS and as specified in 3.1.F.2 and 3.

2. When the reactor is above 70 percent RATED THERMAL POWER with all rods withdrawn, the isothermal temperature coefficient shall be negative, except as specified in 3.1.F.3.

3. If the limits of 3.1.F.1 or 2 cannot be met, POWER OPERATION may continue provided the following actions are taken:

   a. Establish and maintain control rod withdrawal limits sufficient to restore the ITC to less than the limits specified in Specification 3.1.F.1 and 2 above within 24 hours or be in HOT SHUTDOWN within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Figure TS.3.10-2.

   b. Maintain the control rods within the withdrawal limits established above until a subsequent calculation verifies that the ITC has been restored to within its limit for the all rods withdrawn condition.

   c. Submit a special report to the Commission within 30 days, describing the value of the measured ITC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the ITC to within its limit for the all rods withdrawn condition.

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. A reactor shall not be made or maintained critical nor shall the reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.2.C or 3.2.D below):

   1. Two of the three charging pumps shall be OPERABLE.

   2. At least one boric acid tank shall be aligned to the unit and 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.

   3. System piping, valves and pumps 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.

   4. 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.3.

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

3.2.B.6. Motor-operated valve Number 8809C (Boric Acid Storage Tank to the SI Pumps) 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 off position.

7. 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. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit during the time intervals specified, provided STARTUP OPERATION is discontinued until OPERABILITY is restored (except as specified in 3.2.D below). If OPERABILITY is not restored within the time specified, place the affected unit in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

1. Two of the three charging pumps may be inoperable for 72 hours.

2. A unit may operate for 2 hours with no OPERABLE boric acid storage tank.

3. One of the 2 independent flow paths in each unit for boric acid addition to the core may be inoperable for 72 hours. Prior to initiating repairs, the other flow path shall be verified OPERABLE.

4. One channel of heat tracing may be inoperable for 72 hours.

5. Any one redundant automatic valve required for boric acid injection following a steam line break may be inoperable for 72 hours.

6. The valve position monitor light for motor-operated valve No. 8809C (Boric Acid Storage Tank to the SI Pumps) may be inoperable for 72 hours provided the valve position is verified to be open once each shift.

D. During plant shutdown, if the boron concentration of the reactor coolant system is equivalent to or greater than the COLD SHUTDOWN boron concentration, the requirements of 3.2.B.2 are not required to be satisfied.

Correction letter of 3-7-90

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 (except as specified in 3.3.A.2 below):

   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 1270 ±20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of 740 ± 30 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, 3.3.A.4 and 3.1.A.1.d.(2) whenever the reactor coolant system temperature is less than 310°F*.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers are OPERABLE.

*Valid until 20 EFPY

3.3.A.1.f. 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.

g. 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 in the off position.

(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 in the off position.

(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.

2. 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 and 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.

correction letter of 3-7-90

3.3.A.2.g. The valve position monitor lights or alarms for motor-operated valves specified in 3.3.A.1.g above may be inoperable for 72 hours provided the valve position is verified once each shift.

3. At least one safety injection pump control switch in the control room shall be in pullout whenever RCS temperature is less than 310°F* except that both SI pumps may be run while conducting 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.

4. Both safety injection pump control switches in the Control Room shall be in pullout whenever RCS temperature is less than 200°F (except as specified in 3.3.A.3 and 3.1.A.1.d.(2)).

*Valid until 20 EFPY

Correction letter of 3-7-90

3.3.B. <u>Containment Cooling Systems</u>

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 (except as specified in 3.3.B.2 below):

   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. The containment spray system motor operated valves MV-32096 and MV-32097 (Unit 2 valves: MV-32108 and MV-32109) shall be closed and shall have the motor control center supply breakers in the off position.

2. 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 and 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 at least 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.

Correction letter of 3-7-90

3.3.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 (except as specified in 3.3.C.1.b below):

(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.

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 and 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.

3.3.C.2. <u>Two-Unit Operation</u>

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 (except as specified in 3.3.C.2.b below):

(1) Three component cooling pumps are OPERABLE.

(2) Four component cooling heat exchangers 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, place the affected unit(s) in at least HOT SHUTDOWN within the next 6 hours and 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.

3.3.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 (except as specified in 3.3.D.2 below).

   a. Two diesel-driven cooling water pumps are OPERABLE.

   b. Two motor-driven cooling water pumps are OPERABLE.

   c. Two safeguards traveling screens 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 OPERABILITY is not restored within the time specified, 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.

Correction letter of 3-7-90

3.3.D.2.d. One of the Safeguards Traveling Screens may be inoperable for 90 days provided a sluice gate connecting the Emergency Bay and the Circ Water Bay is open (except during periods of testing not to exceed 24 hours).

e. Both Safeguards Traveling Screens may be inoperable for 7 days provided a sluice gate connecting the Emergency Bay and the Circ Water Bay is open.

f. The Emergency Cooling Water line from the Mississippi River may be inoperable for 7 days provided that a sluice gate connecting the Emergency Bay and the Circ Water Bay is open.

Correction letter of 3-7-90

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 the 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. Steam Generator Safety and Power Operated 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 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 6 hours.

1. Ten steam generator safety valves shall be OPERABLE with lift settings of 1077, 1093, 1110, 1120 and 1131 psig ±1% except during testing.

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

B. Auxiliary Feedwater 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 (except as specified in 3.4.B.2 below):

   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.

3.4.B.1.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. Motor operated valves MV-32242 and MV-32243 (Unit 2 valves MV-32248 and MV-32249) 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 in the off position.

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 the time specified, 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 6 hours.

a. A turbine driven AFW pump, system valves and piping may be inoperable for 72 hours.

b. A motor driven AFW pump, system valves and piping may be inoperable for 72 hours.

c. The condensate storage tanks may be inoperable for 48 hours provided the cooling water system is available as a backup supply of water to the auxiliary feedwater pumps.

d. The backup supply of river water provided by the cooling water system may be inoperable for 48 hours provided a minimum of 100,000 gallons of water is available in the condensate storage tanks.

e. The valve position monitor lights for motor operated valves MV-32242 and MV-32243 (Unit 2 valves MV-32248 and MV-32249) may be inoperable for 72 hours provided the associated valves' positions are verified to be open once each shift.

Correction letter of 3-7-90

3.4.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 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 6 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 the 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 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.

TABLE TS.3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 or 2 CANNOT BE MET
1. CONTAINMENT ISOLATION				
a. Safety Injection	(See Item No. 1 of Table TS.3.5-3)			Hot shutdown**
b. Manual	2	1		Hot shutdown
2. CONTAINMENT VENTILATION ISOLATION				
a. Safety Injection	(See Item No. 1 of Table TS.3.5-3)			Maintain Purge and Inservice Purge Valves closed if (1) conditions of a, b, or c cannot be met above COLD SHUTDOWN or (2) if conditions of b or c cannot be met during fuel handling in containment.
b. High Radiation in Exhaust Air	2	1		
c. Manual	2	1		
3. STEAM LINE ISOLATION				
a. Hi-Hi Steam Flow with Safety Injection	2/loop	1		Hot Shutdown**
b. Hi Steam Flow and 2 of 4 Low T_{avg} with Safety Injection	2/loop	1		Hot Shutdown**
c. Hi Containment Pressure	2	1		Hot Shutdown**
d. Manual	1/loop	-		Hot Shutdown**
4. EMERGENCY COOLDOWN EQUIPMENT ROOM ISOLATION				
a. High temperature in ventilation system ducts	2	1		Hot Shutdown**

**If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in COLD SHUTDOWN conditions.

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 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 and 3.6.B.3 below).

2. With one vacuum breaker inoperable with respect to its containment isolation function, apply the requirements of Specification 3.6.C.3, to the isolation valves associated with the inoperable vacuum breaker.

3. One vacuum breaker may be inoperable with respect to its vacuum relief function for 7 days.

C. Containment Isolation Valves

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

2. 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.3 below).

3. 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.

 Correction letter of 3-7-90

3.6.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.

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. The two automatic primary containment isolation valves in each duct that penetrates containment 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. If an inservice purge system automatic primary containment isolation valve or automatic shield building ventilation damper becomes inoperable, apply the requirements of Specification 3.6.C.3.

   e. 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 in-service 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 24 hours initiate the actions 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 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.

3.6.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 (except as specified in 3.F.2 below). 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.

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 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 (except as specified in 3.H.2 below).

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

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.

3.6.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 (except as specified in 3.6.L.2 below).

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

M. Containment Air Locks

1. Each containment air lock shall be OPERABLE with both doors closed whenever CONTAINMENT INTEGRITY is required except as specified in 3.6.M.2 and 3 below; and except for entry and exit, when at least one air lock door shall be closed.

2. With one containment air lock door inoperable:

   a. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

   b. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days (Entry and exit through a closed or locked door is permissible for performance of air lock repairs),

   c. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

3. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

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 (except as specified in 3.7.B below):

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

2. The 4 kV safeguards buses 15 and 16 (Unit 2 buses: 25 and 26) shall be energized.

3. The 480 V safeguards buses 110 and 120 (Unit 2 buses: 210 and 220), and their safeguards motor control centers shall be energized.

4. Reactor protection instrument AC buses shall be energized: 111, 112, 113 and 114 (Unit 2 buses: 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 pumps.

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 (Unit 2 panels: 211, 212, 213 and 214) shall be powered from Panel 117 (Unit 2 panel: 217) or its associated instrument inverter bypass source.

3.7.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, place the affected unit(s) in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

1. 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.1.e within 24 hours**, (b) all engineered safety features equipment associated with the operable diesel generator is OPERABLE, (c) the two required paths from the grid to the plant 4 kV safeguards distribution system are OPERABLE and (d) the OPERABILITY of the two required paths from the grid shall be verified OPERABLE within 1 hour and at least once per 8 hours thereafter.

2. One of the two required paths from the grid to the plant 4 kV safeguards distribution system may be inoperable for 7 days provided (a) D1 and D2 diesel generators are already operating or are demonstrated to be OPERABLE by sequentially performing surveillance requirement 4.6.A.1.e on each diesel generator within 24 hours and (b) the OPERABLE path from the grid shall be verified OPERABLE within 1 hour and at least once per 8 hours thereafter.

3. One of the two required paths from the grid to the plant 4 kV safeguards distribution system 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**, (b) all engineered safety features equipment associated with the OPERABLE diesel generator is OPERABLE, and (c) the OPERABLE path from the grid shall be verified OPERABLE within 1 hour and at least once per 8 hours thereafter.

4. Both of the two required paths from the grid to the plant 4 kV safeguards distribution system may be inoperable for 12 hours provided the D1 and D2 diesel generators are already operating or are demonstrated to be OPERABLE by sequentially performing Surveillance requirement 4.6.A.1.e on each diesel generator within 8 hours.

* The OPERABILITY of the other diesel generator need not be demonstrated if the diesel generator 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.

3.7.B.5. D1 and D2 diesel generators may be inoperable for 2 hours provided the two required paths from the grid to the plant 4 kV safeguards distribution system are OPERABLE and the OPERABILITY of the two required paths from the grid are verified OPERABLE within 1 hour.

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 verified OPERABLE and the diesel generator and safeguards equipment associated with its counterpart are OPERABLE.

7. One battery charger may be inoperable for 8 hours provided, (a) its associated battery is OPERABLE, (b) its redundant counterpart is verified 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 (Unit 2 panels 211, 212, 213 and 214) be powered from an inverter bypass source for 8 hours.

Correction letter of 3-7-90

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 and 3 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 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. The plant shall be in the REFUELING condition.

   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.

correction letter of 3-7-90

3.8.A.1.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 CORE ALTERATIONS.

2. 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.

3. If Specification 3.8.A.1.f or 3.8.A.1.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.

Correction letter of 3-7-90

3.8.B. Fuel Handling and Crane Operation

1. During fuel handling operations or crane operation with loads over spent fuel (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 be 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 fuel handling operations, fuel-handling cranes shall be load-tested for OPERABILITY of limit switches, interlocks and alarms.

   d. 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.

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 spent fuel (inside the spent fuel pool enclosure).

3.8.C. Small Spent Fuel Pool Restrictions

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

D. Spent Fuel Pool Special Ventilation System

1. Both trains of the Spent Fuel Pool Special Ventilation System shall be OPERABLE at all times (except as specified in 3.8.D.2 and 3.8.D.3 below).

2. With one train of the Spent Fuel Pool Special Ventilation System inoperable, fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure) are permissible during the following 7 days, provided the redundant train is demonstrated OPERABLE prior to proceeding with those operations.

3. With both trains of the Spent Fuel Pool Special Ventilation System inoperable, suspend all fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure).

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

E. Storage of Low Burnup Fuel

1. The following restrictions shall apply whenever fuel with an average assembly burnup less than 5,000 MWD/MTU is stored in the spent fuel pool (except as specified in 3.8.E.2 and 3.8.E.3 below):

   a. The boron concentration in the spent fuel pool shall be maintained greater than or equal to 500 ppm, and

   b. Fuel with an average assembly burnup less than 5,000 MWD/MTU shall not be stored in more than three storage locations of every two by two storage rack array.

2. If the conditions in 3.8.E.1.a above are not met, verify that the spent fuel pool storage configuration meets the requirements of specification 3.8.E.1.b and suspend all actions involving the movement of fuel in the spent fuel pool until the boron concentration is increased to 500 ppm or greater.

3. If the conditions in 3.8.E.1.b above are not met, suspend all actions involving movement of fuel in the spent fuel pool, verify the spent fuel pool boron concentration to be greater than or equal to 500 ppm and initiate corrective actions. Mis-positioned fuel assemblies shall be moved to acceptable locations prior to the resumption of other fuel movement in the spent fuel pool.

Correction letter of 3-7-90

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 x 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 $\leq$3.0 mrem to the total body and to $\leq$10 mrem to any organ, and

2. During any calendar year to $\leq$6 mrem to the total body and to $\leq$20 mrem to any organ.

3.9.A.2.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.

3.9.B. <u>Gaseous Effluents</u>

1. <u>Dose Rate</u>

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. <u>Dose from Noble Gases</u>

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.

3.9.B.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 $\leq$15 mrem, and

2. During any calendar year to $\leq$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.

3.9.B.4. 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.

3.9.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 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 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.

3.9.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 Specification 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 Margin

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 changes 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:

   $F_Q^N \times 1.03 \times 1.05 \leq (2.50/P)K(Z)$

   $F_{\Delta H}^N \times 1.04 \leq 1.70 \times [1+ 0.3(1-P)]$

   where the following definitions apply:

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

   - Z is the core height location.

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

3.10.B.1. - F_Q^N or $F_{\Delta H}^N$ is defined as the measured F_Q or $F_{\Delta H}$ respectively, with the smallest margin or greatest excess of limit.

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

- 1.05 is applied to the measured F_Q^N to account for measurement uncertainty.

- 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 difference 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:

$$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (2.50/P) \times K(Z)$$

where V(Z) is defined Figure 3.10-7 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 set-point by 1% for each percent that the measured F_Q^N or by 3.33% for each percent that the measured $F_{\Delta H}^N$ 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 limit.

3.10.B.3. (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 rod 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 target 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.

3.10.B.6. 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. <u>Less than 50 percent of RATED THERMAL POWER:</u>

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

   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. <u>QUADRANT POWER TILT RATIO</u>

1. Except for PHYSICS TESTS, if the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07, 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 QUADRANT POWER TILT RATIO to less than 1.02.

   b. Restrict core power level so as not to exceed RATED THERMAL POWER less 2% for every 0.01 that the QUADRANT POWER TILT RATIO exceeds 1.0.

3.10.C.2. If the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07 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 equivalent 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 control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth control rod prior to this particular low power PHYSICS TEST.

correction letter of 3-7-90

3.10.E. Rod Misalignment Limitations

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

correction letter of 3-7-90

3.10.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 reactor shall be brought to the HOT SHUTDOWN condition within 6 hours should more than one inoperable rod be discovered during POWER OPERATION.

3. If the inoperable rod is located below the 200 step level and is capable of being tripped, or if the 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 rod cannot be located, or if the inoperable 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 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 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.

3.10.I. <u>Monitor Inoperability Requirements</u>

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. <u>DNB Parameters</u>

The following DNB related parameters limits shall be maintained during POWER OPERATION:

a.	Reactor Coolant System Tavg	$\leq$564°F
b.	Pressurizer Pressure	$\geq$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 within the next 4 hours.

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.

*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

Required Shutdown Margin vs Reactor Boron Concentration

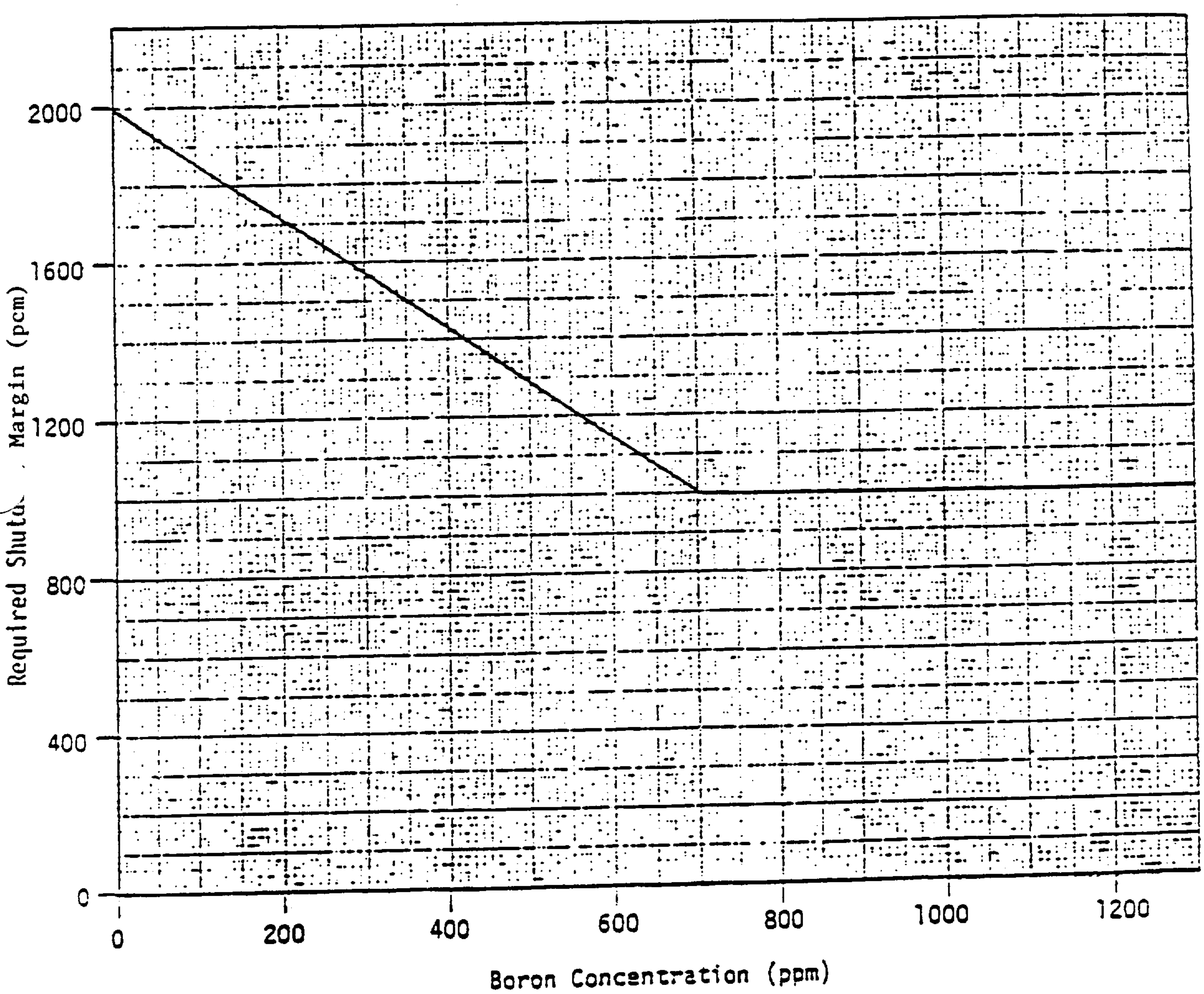

FIGURE TS.3.10-1

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 each refueling so that the power distribution can be confirmed. If the moveable detector system is degraded to the extent that less than 75% of the detector thimbles are available, 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 ALTERATIONS/fuel handling operations are permissible only during the succeeding 7 days. If OPERABILITY is not restored within 7 days, place both units 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.

3.13.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. 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 (except as specified in 3.14.A.2). 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. The system shall be OPERABLE (except as specified in 3.14.B.2 or 3.14.B.3 below) 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

3.14.B.1.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. Spray and Sprinkler Systems

1. Whenever equipment protected by the following spray and sprinkler systems is required to be OPERABLE, the spray and sprinkler system shall be OPERABLE (except as specified in 3.14.C.2 below):

   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 inoperability and the plans for restoring the system to OPERABLE status.

3.14.D. Carbon Dioxide System

1. 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 (except as specified in 3.14.D.3 below).

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.

E. 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 (except as specified in 3.14.E.2 below):

    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.

3.14.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 (except as specified in 3.14.F.2 below):

    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 (except as specified in 3.14.G.2 below).

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 6 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 6 hours.

3.15.B. <u>Radiation Monitors</u>

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.

C. <u>Reactor Vessel Level Instrumentation</u>

1. The reactor vessel level instrumentation channels specified in Table TS.3.15-3 shall be OPERABLE.

2. With the number of OPERABLE reactor vessel level instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-3, either restore the inoperable channels to OPERABLE status within 14 days, or be in at least HOT SHUTDOWN within the next 6 hours.

3. With the number of OPERABLE reactor vessel level instrumentation channels less than the Minimum Channels Operable requirements of Table TS.3.15-3, either restore the minimum number of channels to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours.

Correction letter of 3-7-90

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.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	TEST	FREQUENCY	FSAR Section Reference
14.	Secondary Coolant Gross Beta-Gamma activity	Weekly	
15.	Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)	
16.	Secondary Coolant Chemistry		
	pH	5/week (6)	
	pH Control Additive	5/week (6)	
	Sodium	5/week (6)	

Notes:

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.
2. To determine activity of corrosion products having a half-life greater than 30 minutes.
3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.
4. The maximum interval between analyses shall not exceed 5 days.
5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.
6. The maximum interval between analyses shall not exceed 3 days.

* See Specification 4.1.D

4.4.B. Emergency Charcoal Filter Systems

1. Periodic tests of the Shield Building Ventilation System shall be performed monthly 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 in leakage 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 measurable 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 charcoal adsorbers shall be demonstrated to be less 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).

4.5.B.2. Containment Fan Motors

The Containment Fan Coil Units shall be run on low motor speed for at least 15 minutes at intervals of one month. Motor current shall be measured and compared to the nominal current expected for the test conditions.

3. Valves

a. The refueling water storage tank outlet valves shall be tested in accordance with Section 4.2.

b. The accumulator check valves will be checked for OPERABILITY during each refueling shutdown.

c. The boric acid tank valves to the Safety Injection System shall be tested at intervals of one month.

d. The spray chemical additive tank valves shall be cycled by operator action at intervals of one month.

e. Actuation circuits for Cooling Water System valves that isolate non-essential equipment from the system shall be tested monthly.

f. All motor-operated valves in the SIS, RHR, Containment Spray, Cooling Water, and Component Cooling Water System that are designed for operation during the safety injection or recirculation phase of emergency core cooling, shall be tested for OPERABILITY at each refueling shutdown.

g. The correct position of the throttle valves below shall be verified as follows:

1. Within 4 hours following completion of each valve stroking operation.

2. Within 4 hours following maintenance on the valve when the Safety Injection System is required to be OPERABLE, and

3. Periodically at least once per 18 months to the extent not verified in accordance with 1 and 2 above within this time period.

Unit 1 Valves	Unit 2 Valves
SI-15-6	2SI-15-6
SI-15-7	2SI-15-7
SI-15-8	2SI-15-8
SI-15-9	2SI-15-9

4.5.B.3.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. <u>High Head Safety Injection System</u>:

   (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. <u>RHR System</u>:

   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. This test should be conducted in accordance with the manufacturer's recommendations regarding engine prelube, warm-up, loading and shutdown procedures where possible.

4.6.A.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.

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.

d. This test should be conducted in accordance with the manufacturer's recommendations regarding engine prelube and shutdown procedures where possible.

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. This test should be condutec in accordance with the manufacturer's recommendations regarding engine prelube and shutdown procedures where possible.

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

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

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.

4.6.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.

4.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater, Steam Generator Power Operated Relief Valves, and Steam Exclusion Systems.

Objective

To verify the OPERABILITY of the steam and power conversion systems required for emergency shutdown cooling of the plant.

Specification

A. Auxiliary Feedwater System

1. Each motor-driven auxiliary feedwater pump shall be started at intervals of one month and full flow to the steam generators shall be demonstrated once every refueling shutdown.

2. The steam turbine-driven auxiliary feedwater pump shall be started at intervals of one month* and full flow in the steam generators shall be demonstrated once each year when steam from the steam generators is available.

3. The auxiliary feedwater pumps discharge valves shall be tested by operator action at intervals of one month.

4. Motor-operated valves required to function during accident conditions shall be tested at intervals of one month.

5. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

6. During POWER OPERATION, for the manual valves outside containment, that could reduce auxiliary feedwater flow, if improperly positioned, to less than assumed in the accident analysis, monthly inspections are required to verify the valves are locked in the proper position required for emergency use.

7. After each COLD SHUTDOWN and prior to exceeding 10% power, a test is required to verify the normal flow path from the primary auxiliary feedwater source to the steam generators. This test may consist of maintaining steam generator level during startup with the auxiliary feed pumps.

8. At least once every 18 months during shutdown verify that each pump starts as designed automatically and each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

*If the test comes due during a reactor shutdown the test shall be performed within 24 hours of entering POWER OPERATION.

4.8.B. Steam Generator Power Operated Relief Valves

Each steam generator power operated relief valve shall be isolated and tested monthly.

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 compared monthly 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.

4.11.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. <u>Plant operating records shall be made as follows:</u>

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.

4.13.C. Cont.

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 re-evaluated or the snubber shall be replaced or reconditioned to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, 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 ≥99% DOP removal for particles having a mean diameter of 0.7 microns and ≥99% halogenated hydrocarbon removal.

      b. The results of laboratory carbon sample analysis shall show ≥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 ≥99% DOP removal for particles having a mean diameter of 0.7 microns and ≥99% halogenated hydrocarbon removal.

      b. The results of laboratory carbon sample analysis shall show ≥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.

4.16.B.1.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.

4.17.C. <u>Solid Radioactive Waste</u>

1. <u>Verification of Solidification</u>

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. <u>Dose from All Uranium Fuel Cycle Sources</u>

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 surveillance 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.

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

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

References

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

6.7 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 totaling 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.

6.7.A.1.c cont.

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

2. Startup Report

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

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

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.

6.7.A.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.

6.7.A.5. Annual Summaries of Meteorological Data

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

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

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

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

2. <u>Environmental Special Reports</u>

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

3. <u>Other Environmental Reports (non-radiological, non-aquatic)</u>

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

a. Environmental events that indicate or could result in a significant environmental impact casually related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by the Endangered Species Act of 1973; or increase in nuisance organisms or conditions. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their prelimiary responses.

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

D. <u>Special Reports</u>

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

BASES FOR

SECTION 2.0

SAFETY LIMITS AND LIMITING SAFETY

SYSTEM SETTING

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 and WRB-1 DNB correlations. The W-3 DNB correlation is used for Exxon fuel. The WRB-1 DNB correlation is used for Westinghouse fuel. The W-3 and WRB-1 DNB correlations have 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 for the Exxon Nuclear fuel and to 1.17 for the Westinghouse fuel. These limits correspond 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.

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 $\sim\Delta T$).

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 are evaluated using standard DNB methodology. For all four curves the DNBR is limiting 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 conservative for the following nuclear hot channel factors:

$$F^N_{\Delta H} = 1.70\ [1 + 0.3(1-P)]\ ;\ \text{and}\ F^N_Q = 2.50$$

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10.

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 for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

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, the reactor high pressure trip and the safety valves 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.

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, $\geq$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 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 drop events 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 14.3
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

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 operators to place the plant in "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 and in COLD SHUTDOWN within the following 30 hours. The action statement provides up to 108 hours (72 + 6 + 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 48 hours (72 - 60 = 12, 12 + 6 + 30 = 48).

2. However, if this condition were discovered while in HOT SHUTDOWN, the unit could remain in HOT SHUTDOWN for the next 72 hours. 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 could not be used since the plant is already below that condition.

The above paragraphs apply to all Section 3 requirements.

Correction letter of 3-7-90

3.1 REACTOR COOLANT SYSTEM

Bases continued

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 loops 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 for Exxon fuel and 1.17 for Westinghouse fuel. 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.

Two methods of removing decay heat are required at all times except during REFUELING. Above 350°F, both reactor coolant loops must be OPERABLE to serve this function. Below 350°F a reactor coolant loop or a residual heat removal loop is capable of removing decay heat and any combination of two loops serve this function.

Specification 3.1.A.1.d.(2) allows the use of one safety injection pump to ensure that adequate core cooling and reactor coolant system inventory can be maintained in the event of a loss of Residual Heat Removal System cooling during reduced inventory conditions. A reduced inventory condition, as defined by Generic Letter 88-17, Loss of Decay Heat Removal, exists whenever the reactor vessel water level is lower than three feet below the reactor vessel flange. The operation of a safety injection pump under such conditions would be controlled by an approved emergency operating procedure.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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 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.

The minimum pressurization temperatue (310°F *) is determined from Figure TS.3.1-1 and is the temperature equivalent to the RCS safety relief valve setpoint pressure. The RCS safety valves and normal setpoints on the pressurizer PORV's do not provide overpressure protection for certain low temperature operational transients. Inadvertent pressurization of the RCS at temperatures below 310°F* could result in the limits of Figures TS.3.1-1 and TS.3.1-2 being exceeded. Thus the low temperature overpressure mitigating system, which is designed to prevent pressurizing the RCS above the pressure limits specified in Figures TS.3.1-1 and TS.3.1-2, is enabled at 310°F*. Above 310°F* the RCS safety valves would limit the pressure increase and would prevent the limits of Figures TS.3.1-1 and TS.3.1-2 from being exceeded.

*Valid until 20 EFPY

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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.

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

Appendix G of 10 CFR Part 50, and the ASME Code require that the reactor coolant pressure boundary be designed with sufficient margin to insure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner, the probability of rapidly propagating fracture is minimized and the design reflects the uncertainties in determining the effects of irradiation on material properties. Figures TS.3.1-1 and 2 have been developed (Reference 1) in accordance with these regulations. The curves are based on the properties of the most limiting material in either unit's reactor vessel (Unit 1 reactor vessel weld W-3) and are effective to 20 EFPY. The curves have been adjusted for possible errors in the pressure and temperature sensing instruments.

The curves define a region where brittle fracture will not occur and are determined from the material characteristics, irradiation effects, pressure stresses and stresses due to thermal gradients across the vessel wall.

Heatup Curves

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. At the inner wall of the vessel, the thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, 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 inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. For the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis. The heatup limit curve is a composite curve prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour.

Cooldown Curves

During cooldown, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from tensile at the inner wall to compressive at the outer wall. The thermal induced tensile stresses at the inner wall are additive to the pressure induced tensile stresses which are already present. Therefore, the controlling location is always the inside wall.

3.1 REACTOR COOLANT SYSTEM

Bases (continued)

The cooldown limit curves were prepared utilizing the same type of analysis used to calculate the heatup curve except that the controlling location is always the inside wall.

Limit lines for cooldown rates between those presented may be obtained by interpolation.

Criticality Limits

Appendix G of 10 CFR Part 50 requires that for a given pressure, the reactor must not be made critical unless the temperature of the reactor vessel is 40°F above the minimum permissible temperature specified on the heatup curve and above the minimum permissible temperature for the inservice hydrostatic pressure test. For Prairie Island the curves were prepared, requiring that criticality must occur above the maximum permissible temperature for the inservice hydrostatic pressure test.

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.

ASME Code Section XI Inservice Test Limits

The pressure temperature limits for the ASME Code Section XI Inservice Test Limits (hydrostatic pressure test) are less restrictive than the heatup and cooldown curves to allow for the periodic inservice hydrostatic test. These limits are allowed to be less restrictive because the hydrostatic test is based on a 1.5 safety factor versus the 2.0 safety factor built into the heatup and cooldown curves and because the test is run at a constant temperature so the thermal stresses in the vessel are minimal.

Steam Generator Pressure/Temperature Limitations

The limitations on steam generator pressure and temperature ensure that the pressure induced stress in the steam generators do not exceed the maximum allowable fracture toughness stress limits and thus prevent brittle fracture of the steam generator shell.

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 in accordance with ASME Code requirements.

Reference

1. USAR Section 4.2

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 based 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 specifies 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.3.1-3, 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.3.1-3 should be minimized since the activity levels allowed by Figure TS.3.1-3 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 tube 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 concentrations to exist for 48 hours for coolant temperatures less than 250°F and still provide the assurance the integrity of the primary coolant system will be maintained.

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

References

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

3.1 REACTOR COOLANT SYSTEM

Bases continued

F. Isothermal Temperature Coefficient (ITC)

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 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 low power PHYSICS TESTS in order to verify analytical prediction. The units of the isothermal temperature coefficient are pcm/°F, where 1pcm = $1 \times 10^{-5}\ \Delta k/k$,

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 isothermal temperature coefficient could exist at beginning of cycle (BOC). Safety analyses verify the acceptability of the isothermal temperature coefficient for limits specified in 3.1.F. 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 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 (Reference 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. One boric acid tank must be aligned to each unit whose temperature exceeds 200°F (unless the plant is shutting down and the necessary boric acid has been injected). 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.

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:

3.3 ENGINEERED SAFETY FEATURES

Bases continued

(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 THERMAL 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.

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

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.

Maintaining both safety injection pump Control Room control switches in pullout, as specified in 3.3.A.4, will ensure that the RCS pressure/temperature limitations specified in Figures TS.3.1-1 and TS.3.1-2 will not be exceeded, at low RCS temperatures, as the result of mass input into the RCS from an inadvertent safety injection pump start.

3.3 ENGINEERED SAFETY FEATURES

Bases continued

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 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 Safeguards Traveling Screens and Emergency Cooling Water Supply line are designed to provide a supply of screened cooling water in the event that an earthquake 1) destroys Dam No. 3 (dropping the water level in the normal canal to the screenhouse) and 2) causes the banks bordering the normal canal to the screenhouse to collapse eliminating the river as a source of cooling water. The Safeguards Traveling Screens and Emergency Cooling Water Supply line provide an alternate supply of water to the Safeguards Bay, which contains the two diesel driven and the one vertical motor driven cooling water pumps. Their normal supply is from the Circ

3.3 ENGINEERED SAFETY FEATURES

Bases continued

Water Bay thru one of two sluice gates. Either one of the two sluice gates or one of the two Safeguards Traveling Screens will adequately supply any of the three cooling water pumps. The Safeguards Traveling Screens are not considered part of the "engineered safety features associated with the operable diesel-driven cooling water pump" for determination of operability of diesel-driven cooling water pumps.

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 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.

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 steam generator 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 steam generator 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 steam generator 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. During STARTUP OPERATIONS, the Auxiliary Feedwater motor-operated injection valves maybe less than full open as necessary to faciliate plant startup.

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 steam generator 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 break within the containment, by the high containment pressure signal.

The safety injection 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 set-point 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 minor 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 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.

Air locks are provided with two doors, each of which is designed to seal against the maximum containment pressure resulting from the limiting DBA. Should an air lock become inoperable as a result of an inoperable air lock door or an inoperable door interlock, power operation may continue provided that at least one OPERABLE air lock door is closed. With an air lock door inoperable, access through the closed or locked OPERABLE door is only permitted for repair of inoperable air lock equipment.

OPERABILITY of air locks is required to ensure that CONTAINMENT INTEGRITY maintained. Should an air lock become inoperable for reasons other than an inoperable air lock door, the air lock leak tight integrity must be restored within 24 hours or actions must be taken to place the unit in a condition for which CONTAINMENT INTEGRITY is not required.

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. Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. 3.8.A.1.h 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 10 CFR 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

Correction letter of 3-7-90

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. 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.

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.

The requirements for the storage of low burnup fuel in the spent fuel pool ensure that the spent fuel pool will remain subcritical during fuel storage. Fuel stored in the spent fuel pool will be limited to a maximum enrichment of 4.25 weight percent U-235. It has been shown by criticality analysis that the use of the three out of four storage configuration will assure that the K_{eff} will remain less than 0.95, including uncertainties, when fuel with a maximum enrichment of 4.25 weight percent U-235 and average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool.

The requirement for maintaining the spent fuel pool boron concentration greater than 500 ppm whenever fuel with average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool ensures that K_{eff} for the spent fuel pool will remain less than 0.95, including uncertainties, even if a fuel assembly is inadvertently inserted in the empty cell of the three out of four storage configuration.

References

1. USAR, Section 10.2.1.2
2. USAR, Section 14.5.1
3. USAR, Section 10.3.7
4. Exhibit C, NSP License Amendment Request Dated December 21, 1984.

3.9 RADIOACTIVE EFFLUENTS

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 Condition 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 low 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 RADIOACTIVE EFFLUENTS

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 RADIOACTIVE EFFLUENTS

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 no 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 <u>RADIOACTIVE EFFLUENTS</u>

<u>Bases</u> 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 RADIOACTIVE EFFLUENTS

Bases continued

D. Dose From All Uranium Fuel Cycle Sources

This specification is provided 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 Margin

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown margin 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 for Exxon fuel and 1.17 for Westinghouse fuel 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. The ECCS analysis was performed in accordance with SECY 83-472. One calculation at the 95% probability level was performed as well as one calculation with all the required features of 10 CFR Part 50, Appendix K. The 95% probability level calculation used a peak linear heat generation rate of 14.2 kw/ft. The Appendix K calculation used a peak linear heat generation rate of 15.8 kw/ft for the F_Q limit of 2.5. Maintaining 1) peaking factors below the F_Q limit of 2.5 during all Condition I events and 2) the peak linear heat generation rate below 14.2 kw/ft at the 95% probability level assures compliance with the ECCS analysis.

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 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^N is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux on the surface of a fuel rod divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

The K(Z) function shown in Figure TS.3.10-5 is a normalized function that limits F_Q axially. The K(Z) specified for the lowest six (6) feet of the core is arbitrarily 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.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

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 variations 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_Q^N (equil) is the measured limiting F_Q^N obtained at equilibrium conditions during target flux determination.

$F_{\Delta H}^N$, 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.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

When a measurement of $F^N_{\Delta H}$ is taken, measurement 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.

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 offset which is defined as the difference in normalized power between the top and bottom halves of the core.

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 peaking factors. 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 for Exxon fuel and 1.17 for Westinghouse fuel 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 RATIO

QUADRANT POWER TILT RATIO 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 RATIO (continued)

core limits protected per Specification 3.10.E. A quadrant power tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant power 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 QUADRANT POWER TILT RATIOS 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 THERMAL 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 QUADRANT POWER TILT RATIO of 1.02 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 RATIO.

An increase in F_Q^N is not likely to occur with QUADRANT POWER TILT RATIOS up to 1.03 because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q^N occurs.

Therefore, a limiting QUADRANT POWER TILT RATIO of 1.03 can be tolerated. However, a measurement uncertainty is associated with the indicated QUADRANT POWER TILT RATIO. Thus, allowing for a low measurement of QUADRANT POWER TILT RATIO, the action level of indicated QUADRANT POWER TILT RATIO has been set at 1.02. An alarm is set to alert the operator to an indicated QUADRANT POWER TILT RATIO of 1.02 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 0.01 of QUADRANT POWER TILT RATIO 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 RATIO (continued)

The upper limit on the QUADRANT POWER TILT RATIO 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 QUADRANT POWER TILT RATIO is restricted below 1.07.

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.

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 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. The fire 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 <u>FIRE DETECTION AND PROTECTION SYSTEMS</u>

<u>Bases</u> 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 continuous 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.

Core exit thermocouple readings necessary to meet the requirements of Specification 3.15.A are available from the Plant Process Computer, the Control Room Core Exit Thermocouple Display or if no other readout is available, from test equipment readings from the Core Exit Thermocouple Junction Boxes.