Vepco

SURRY POWER STATION
UNITS 1 AND 2

TECHNICAL SPECIFICATIONS

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

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1.0 DEFINITIONS

The following frequently used terms are defined for the uniform interpretation of the specifications.

A. RATED POWER

A steady state reactor core heat output of 2546 MWt.

B. THERMAL POWER

The total core heat transferred from the fuel to the coolant.

C. REACTOR OPERATION

1. REFUELING SHUTDOWN

When the reactor is subcritical by at least 5% $\Delta k/k$ and T_{avg} is $\leq$140°F and fuel is scheduled to be moved to or from the reactor core.

2. COLD SHUTDOWN

When the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is $\leq$200°F.

3. INTERMEDIATE SHUTDOWN

When the reactor is subcritical by at least 1.77% $\Delta k/k$ and 200°F < T_{avg} < 547°F.

4. HOT SHUTDOWN

When the reactor is subcritical by at least 1.77% $\Delta k/k$ and T_{avg} is $\geq$ 547°F.

5. REACTOR CRITICAL

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

6. POWER OPERATION

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

7. REFUELING OPERATION

Any operation involving movement of core components when the vessel head is unbolted or removed.

D. OPERABLE

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). The system or component shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Section 3, and (2) it has been tested periodically in accordance with Section 4 and meets its performance requirements.

E. PROTECTIVE INSTRUMENTATION LOGIC

1. ANALOG CHANNEL

An arrangement of components and modules as required to generate a single protective action digital signal when required by a unit condition. An analog channel loses its identity when single action signals are combined.

2. <u>AUTOMATIC ACTUATION LOGIC</u>

A group of matrixed relay contacts which operate in response to the digital output signals from the analog channels to generate a protective action signal.

F. <u>INSTRUMENTATION SURVEILLANCE</u>

1. <u>CHANNEL CHECK</u>

The qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation on channels measuring the same parameter.

2. <u>CHANNEL FUNCTIONAL TEST</u>

Injection of a simulated signal into an analog channel as close to the sensor as practicable or makeup of the logic combinations in a logic channel to verify that it is operable, including alarm and/or trip initiating action.

3. <u>CHANNEL CALIBRATION</u>

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

G. <u>CONTAINMENT INTEGRITY</u>

Containment integrity shall exist when:

a. The penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by at least one closed manual valve, blind flange, or deactivated automatic valve secured in its closed position except as provided in Specification 3.8.C. Non-automatic or deactivated automatic containment isolation valves may be opened intermittently for operational activities provided that the valves are under administrative control and are capable of being closed immediately, if required.

b. The equipment access hatch is closed and sealed.

c. Each airlock is OPERABLE except as provided in Specification 3.8.B.

d. The containment leakage rates are within the limits of Specification 4.4.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

H. REPORTABLE EVENT

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

I. QUADRANT POWER TILT

The quadrant power tilt is defined as the ratio of the maximum upper excore detector current to the average of the upper excore detector currents or the ratio of the maximum lower excore detector current to the average of the lower excore detector currents whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

J. LOW POWER PHYSICS TESTS

Low power physics tests conducted below 5% of rated power which measure fundamental characteristics of the core and related instrumentation.

K. FIRE SUPPRESSION WATER SYSTEM

A fire suppression water system shall consist of: a water source(s), gravity tank(s) or pump(s), and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

L. OFFSITE DOSE CALCULATION MANUAL (ODCM)

The Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.6.B.2 and 6.6.B.3.

M. DOSE EQUIVALENT I-131

The dose equivalent I-131 shall be that concentration of I-131 (microcurie/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" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

N. GASEOUS RADWASTE TREATMENT SYSTEM

A gaseous radwaste treatment system is 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.

O. PROCESS CONTROL PROGRAM (PCP)

The process control program shall contain the current formula, sampling, analyses, 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, State regulations, and other requirements governing the disposal of the waste.

P. PURGE - PURGING

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

Q. VENTILATION EXHAUST TREATMENT SYSTEM

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents. Treatment includes 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 (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

R. VENTING

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

S. SITE BOUNDARY

The site boundary shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

T. UNRESTRICTED AREA

An unrestricted area shall be any area at or beyond the site boundary where access is not controlled by the licensee for purpose of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential qauarters or for industrial, commerical, institutional, or recreational purposes.

U. MEMBER(S) OF THE PUBLIC

Member(s) of the public shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

V. CORE OPERATING LIMITS REPORT

The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.2.C. Plant operation within these limits is addressed in individual specifications.

W. STAGGERED TEST BASIS

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature and coolant flow when a reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

A. The combination of reactor thermal power level, coolant pressure, and coolant temperature shall not:

1. Exceed the limits shown in TS Figure 2.1-1 when full flow from three reactor coolant pumps exist.
2. Exceed the limits shown in TS Figure 2.1-2 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are open.
3. Exceed the limits shown in TS Figure 2.1-3 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are closed.

4. The reactor thermal power level shall not exceed 118% of rated power.

B. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of the rated power.

Basis

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 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 reactor 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, DNB has been correlated to thermal power, reactor coolant temperature and reactor coolant pressure which are observable parameters. This correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the DNB heat flux at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB basis is as follows: there must be at least a 95% probability with 95% confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is based on the entire applicable experimental data set to meet this statistical criterion.(1)

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the calculated DNBR is not less than the design DNBR limit or the average enthalpy at the exit of the vessel is equal to the saturation value. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon the design DNBR limit alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The effects of rod bowing are also considered in the DNBR analyses.

TS Figure 2.1-1 is based on a 1.55 cosine axial flux shape and a statistical treatment of key DNBR analysis parameter uncertainties including an enthalpy rise hot channel factor which follows the following functional form: $F\Delta H(N) = 1.56 [1 + 0.3(1-P)]$ where P is the fraction of RATED POWER. The limits include margin to accommodate rod bowing.(1) TS Figures 2.1-2 and 2.1-3 are based on an FΔH(N) of 1.55, a deterministic treatment of key DNB analysis parameter uncertainties, and include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor. The FΔH(N) limit presented in the unit- and reload-specific CORE OPERATING LIMITS REPORT is confirmed for each reload to be accommodated by the Reactor Core Safety Limits.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits as specified in the CORE OPERATING LIMITS REPORT ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 573.0°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

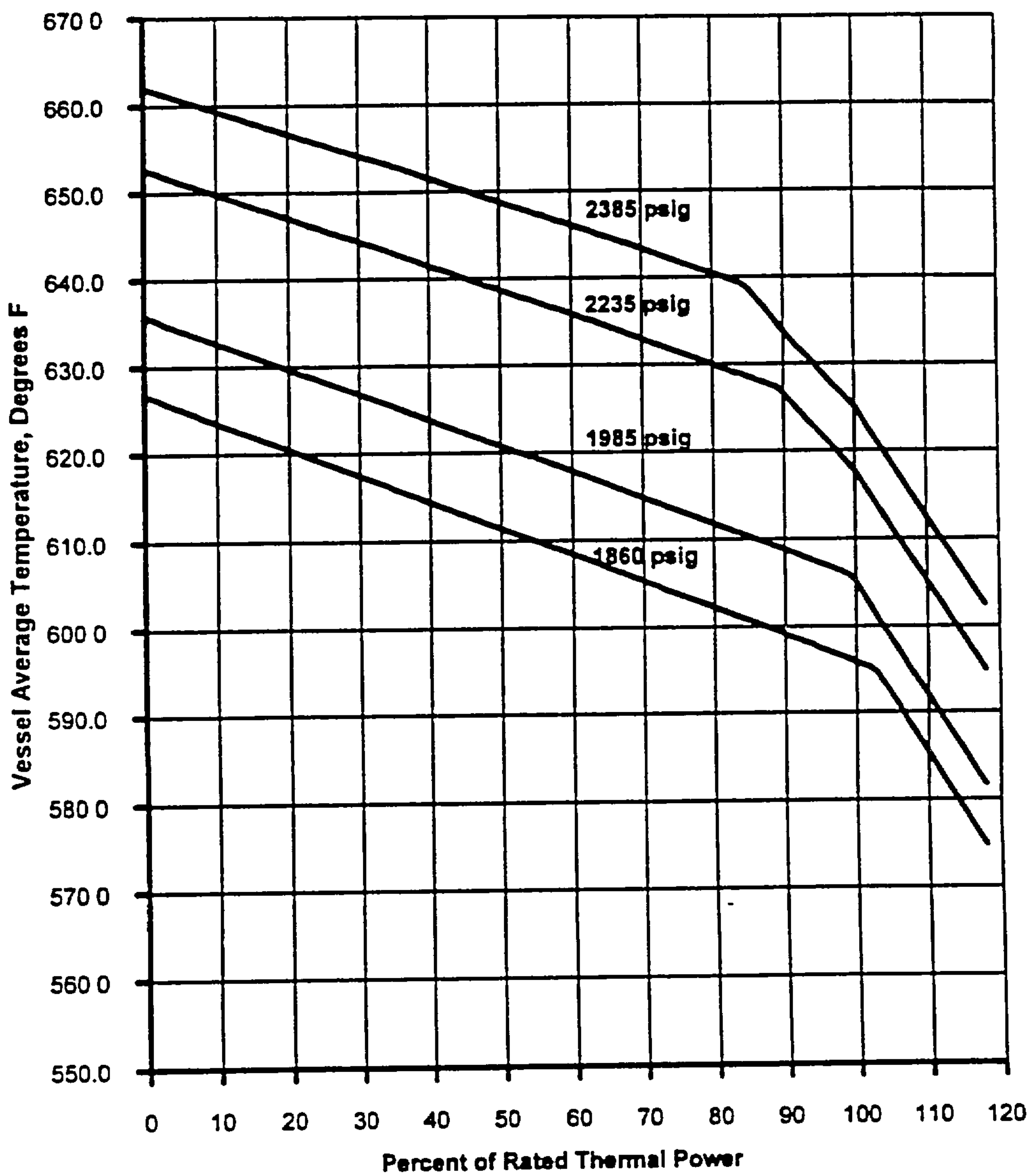
For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also ensures that at least 99.9% of the core avoids the onset of DNB when the limiting rod is at the DNBR limit.

The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

References

1) FSAR Section 3.4
2) FSAR Section 3.3
3) FSAR Section 14.2

TS FIGURE 2.1-1
REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS
THREE LOOP OPERATION, 100% FLOW



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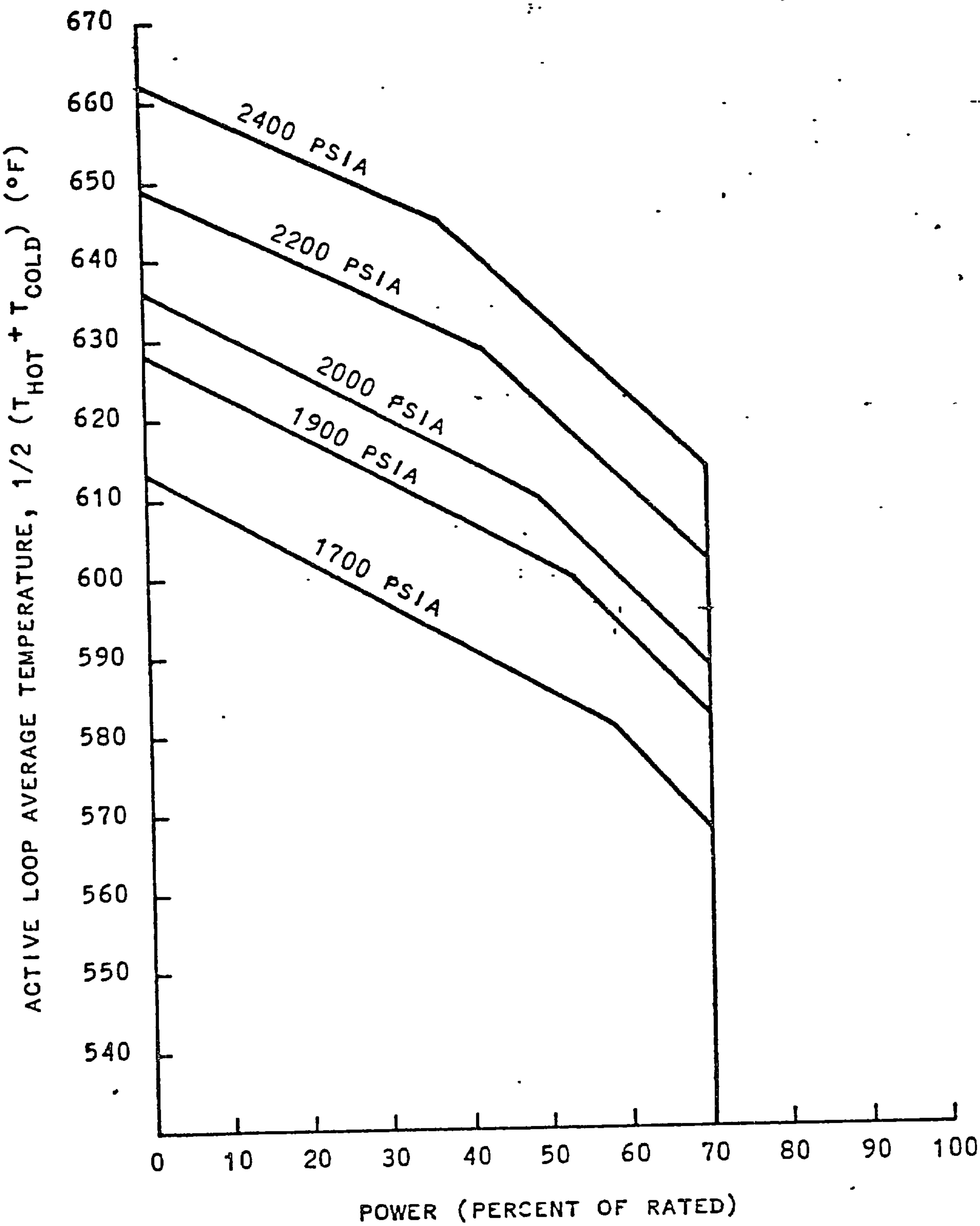


FIGURE 2.1-2 REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES OPEN

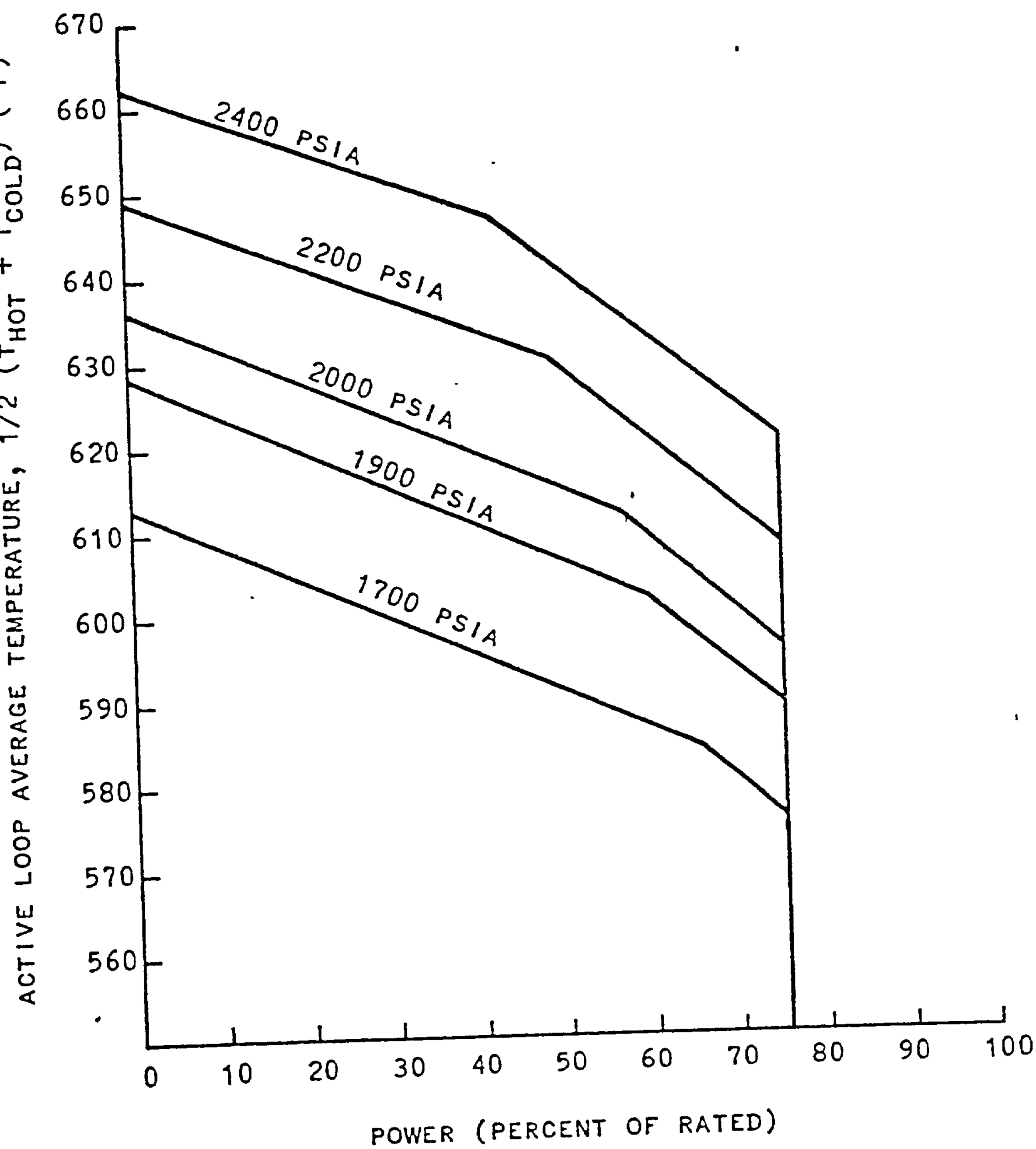
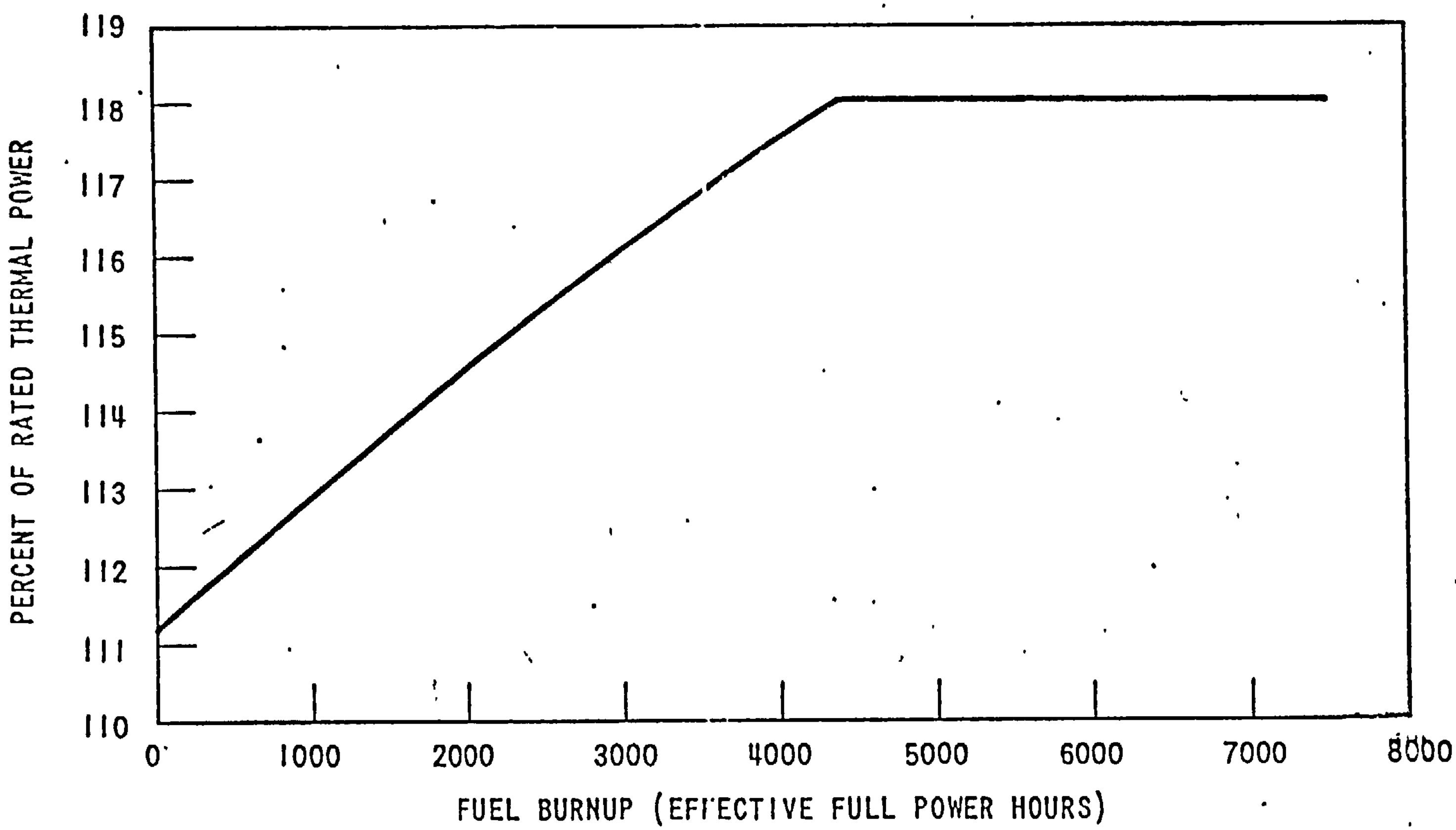


FIGURE 2.1-3 REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES CLOSED

CHANGE NO. 6



CHANGE NO. 9

Figure 2.1-4. Thermal Overpower Limit

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System pressure.

Objective

To maintain the integrity of the Reactor Coolant System.

Specification

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis

The Reactor Coolant System(1) serves as a barrier which prevents radionuclides contained in the reactor coolant from reaching the environment. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the release of fission products. 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.(2)

The nominal settings of the power-operated relief valves at 2335 psig, the reactor high pressure trip at 2385 psig and the safety valves at 2485 psig are established to assure never reaching the Reactor Coolant System pressure safety limit. The initial hydrostatic test has been conducted at 3107 psig to assure the integrity of the Reactor Coolant System.

1) UFSAR Section 4
2) UFSAR Section 4.3

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip and permissive settings for instruments monitoring reactor power; and reactor coolant pressure, temperature, and 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, power range (low set point) - $\leq$ 25% of rated power.

(b) High flux, intermediate range (high set point) - current equivalent to $\leq$ 40% of full power.

(c) High flux, source range (high set point) - Neutron flux $\leq 10^6$ counts/sec.

2. Core Protection

(a) High flux, power range (high set point) - $\leq$ 109% of rated power.

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

(c) Low pressurizer pressure - $\geq$ 1860 psig.

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_0 [K_1 - K_2 (\frac{1 + t_1 S}{1 + t_2 S}) (T - T') + K_3 (P - P') - f(\Delta I)]$$

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = 573.0°F

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.135

K_2 = 0.01072

K_3 = 0.000566

$\Delta I = q_t - q_b$, where q_t 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

$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1

t_1 = 25 seconds

t_2 = 3 seconds

(e) Overpower ΔT

$$\Delta T \leq \Delta T_0 [K_4 - K_5 (\frac{t_3 S}{1 + t_3 S}) T - K_6 (T - T') - f(\Delta I)]$$

where

ΔT_0 = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = Average coolant temperature measured at nominal conditions and rated power, °F

K_4 = A constant = 1.089

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature 0.02/°F

K_6 = 0 for $T \leq T'$

= 0.001086 for $T > T'$

$f(\Delta I)$ as defined in (d) above,

τ_3 = 10 seconds

(f) Low reactor coolant loop flow = ≥ 90% of normal indicated loop flow as measured at elbow taps in each loop

(g) Low reactor coolant pump motor frequency - ≥ 57.5 Hz

(h) Reactor coolant pump under voltage - ≥ 70% of normal voltage

3. Other reactor trip settings

(a) High pressurizer water level - ≤ 92% of span

(b) Low-low steam generator water level - ≥ 14.5% of narrow range instrument span

(c) Low steam generator water level - ≥ 15% of narrow range instrument span in coincidence with steam/feedwater mismatch flow - ≤ 1.0×10^6 lbs/hr

(d) Turbine trip

(e) Safety injection - Trip settings for Safety Injection are detailed in TS Section 3.7.

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

1. The reactor trip on low pressurizer pressure, high pressurizer level, turbine trip, and low reactor coolant flow for two or more loops shall be unblocked when power ≥ 10% of rated power.

2. The single loop loss of flow reactor trip shall be unblocked when the power range nuclear flux ≥ 50% of rated power.

3. The power range high flux, low setpoint trip and the intermediate range high flux, high setpoint trip shall be unblocked when power ≤ 10% of rated power.

4. The source range high flux, high setpoint trip shall be unblocked when the intermediate range nuclear flux is ≤ 5×10^{-11} amperes.

Basis

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip value was used in the safety analysis.[1] The Source Range High Flux Trip provides reactor core protection during shutdown (COLD SHUTDOWN, INTERMEDIATE SHUTDOWN, and HOT SHUTDOWN) when the reactor trip breakers are closed and reactor power is below the permissive P-6. The Source and Intermediate Range trips in addition to the Power Range trips provide core protection during reactor startup when the reactor is critical. The Source Range channels will initiate a reactor trip at about 10^6 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a reactor trip at a current level proportional to ≤ 40% of RATED POWER unless manually blocked when P-10 becomes active. In the accident analyses, bounding transient analysis results are based on reactivity excursions from an initially critical condition, where the Source Range trip is assumed to be blocked. Accidents initiated form a subcritical condition would produce less severe results, since the Source Range trip would provide core protection at a lower power level. No credit is taken for operation of the Intermediate Range High Flux trip. However, its functional capability is required by this specification to enhance the overall reliability of the Reactor Protection System.

The high and low pressurizer pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss-of-coolant accident.(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 the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3 seconds), and pressure is within the range between high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,(2) is always below the core safety limit as shown on TS Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor limit is automatically reduced.(4)(5)

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification on core safety limits and to apply to 100% of design flow. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as shown in Figures 2.1-1 through 2.1-3. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The overpower protection system set points include the effects of fuel densification. | 21

In order to operate with a reactor coolant loop out of service (two-loop operation) and with the stop valves of the inactive loop either open or closed, the overtemperature ΔT trip setpoint calculation has to be modified by the adjustment of the variable K_1. This adjustment, based on limits of two-loop operation, provides sufficient margin to DNB for the aforementioned transients during two loop operation. The required adjustment and subsequent mandatory calibrations are made in the protective system racks by qualified technicians* in the same manner as adjustments before initial startup and normal calibrations for three-loop operation. | 21

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed Section 7 and specified in Section 14.2.2 of the FSAR 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 setpoints meet this requirement and include allowance for instrument errors.(2)

*As used here, a qualified technician means a technician who meets the requirements of ANS-3. He shall have a minimum of two years of working experience in his speciality and at least one year of related technical training.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The undervoltage reactor trip protects against a decrease in Reactor Coolant System flow caused by a loss of voltage to the reactor coolant pump busses. The underfrequency reactor trip (opens RCP supply breakers and) protects against a decrease in Reactor Coolant System flow caused by a frequency decay on the reactor coolant pump busses. The undervoltage and underfrequency reactor trips are expected to occur prior to the low flow trip setpoint being reached for low flow events caused by undervoltage or underfrequency, respectively. The accident analysis conservatively ignores the undervoltage and underfrequency trips and assumes reactor protection is provided by the low flow trip. The undervoltage and underfrequency reactor trips are retained as back-up protection.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1154 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error(7) and transient level overshoot beyond this 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 setpoint 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.(7)

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal unit operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two or more reactor coolant pumps are lost. Above 50%, an automatic reactor trip will occur if any pump is lost or de-energized. This latter trip will prevent the minimum value of the DNBR from going below the applicable design as a result of the decrease in Reactor Coolant System flow associated with the loss of a single reactor coolant pump.

Although not necessary for core protection, other reactor trips provide additional protection. The steam/feedwater flow mismatch which is coincident with a low steam generator water level is designed for and provides protection from a sudden loss of the reactor's heat sink. Upon the actuation of the safety injection circuitry, the reactor is tripped to decrease the severity of the accident condition. Upon turbine trip, at greater than 10% power, the reactor is tripped to reduce the severity of the ensuing transient.

References

(1) FSAR Section 14.2.1
(2) FSAR Section 14.2
(3) FSAR Section 14.5
(4) FSAR Section 7.2
(5) FSAR Section 3.2.2
(6) FSAR Section 14.2.9
(7) FSAR Section 7.2

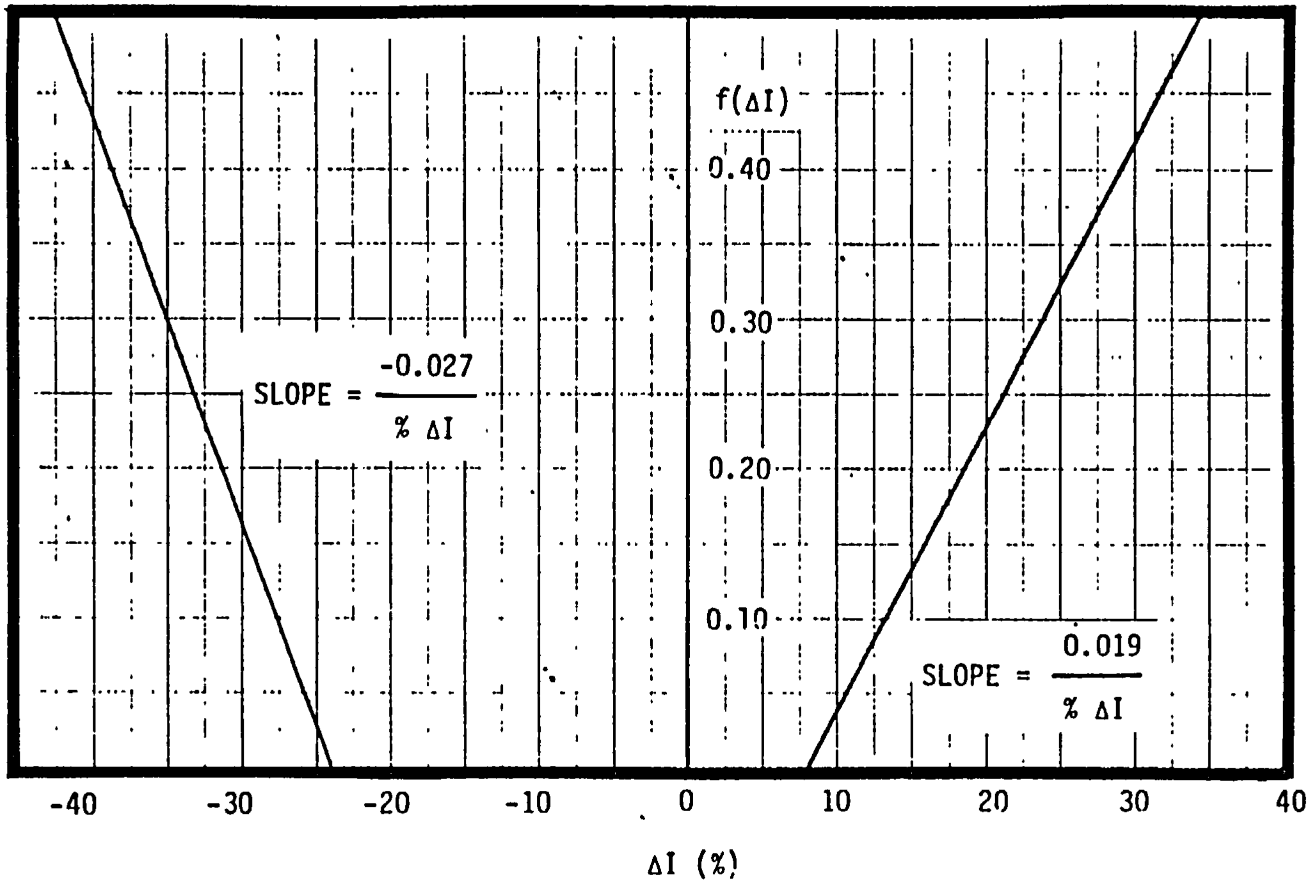


Figure 2.3-1 OPΔT and OTΔT f(ΔI) Function

3.0 LIMITING CONDITIONS FOR OPERATION

3.0.1 In the event a Limiting Condition for Operation and/or associated modified requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within 6 hours and in at least cold shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

3.0.2 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 specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least hot shutdown within 6 hours and in at least cold shutdown within the following 30 hours. This specification is not applicable in cold shutdown or refueling shutdown conditions.

Basis

3.0.1 This specification delineates the action to be taken for circumstances not directly provided for in the action statements and whose occurrence would violate the intent of the specification. For example, Specification 3.3 requires each Reactor Coolant System accumulator to be operable and provides explicit action requirements if one accumulator is inoperable. Under the terms of Specification 3.0.1, if more than one accumulator is inoperable, the unit is required to be in at least hot shutdown within 6 hours. As a further example, Specification 3.4 requires two Containment Spray Subsystems to be operable and provides explicit action requirements if one spray system is inoperable. Under the terms of Specification 3.0.1, if both of the required Containment Spray Subsystems are inoperable, the unit is required to be in at least hot shutdown within 6 hours and in at least cold shutdown in the next 30 hours. It is assumed that the unit is brought to the required condition within the required times by promptly initiating and carrying out the appropriate action.

3.0.2 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the actions for power sources, when a normal or emergency power source is not operable. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the action statements associated with individual systems, subsystems, trains, components or devices to be consistent with the action statements of the associated electrical power source. It allows operation to be governed by the time limits of the action statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual action statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.16 requires in part that two emergency diesel generators be operable. The action statement provides for out-of-service time when one emergency diesel generator is not operable. If the definition of operable were applied without consideration of Specification 3.0.2, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.2 permit the time limits for continued operation to be consistent with the action statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be operable, and all redundant systems, subsystems, trains, components and devices must be operable, or otherwise satisfy Specification 3.0.2 (i.e., be capable of performing their design function and have at least one normal or one emergency power source operable). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.16 requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be operable. The action statement provides out-of-service time when one required offsite circuit is not operable. If the definition of operable were applied without consideration of Specification 3.0.2, all systems, subsystems, trains, components and devices supplied by the inoperable normal power source, one of the offsite circuits, would be inoperable. This would dictate invoking the applicable action statements for each of the applicable LCOs. However, the provisions of Specification 3.0.2 permit the time limits for continued operation to be consistent with the action statement for the inoperable normal power source instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be operable (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be operable, or likewise satisfy Specification 3.0.2 (i.e., be capable of performing their design functions and have an emergency power source operable). In other words, both emergency power sources must be operable and all redundant systems, subsystems, trains, components and devices in both divisions must also be operable. If these conditions are not satisfied, shutdown is required in accordance with this specification.

In cold shutdown or refueling shutdown conditions, Specification 3.0.2 is not applicable, and thus the individual action statements for each applicable Limiting Condition for Operation in these conditions must be adhered to.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objectives

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe REACTOR OPERATION.

These conditions relate to: operational components, heatup and cooldown, leakage, reactor coolant activity, oxygen and chloride concentrations, minimum temperature for criticality, and Reactor Coolant System overpressure mitigation.

A. Operational Components

Specifications

1. Reactor Coolant Pumps

   a. A reactor shall not be brought critical with less than three pumps, in non-isolated loops, in operation.

b. If an unscheduled loss of one or more reactor coolant pumps occurs while operating below 10% RATED POWER (P-7) and results in less than two pumps in service, the affected plant shall be shutdown and the reactor made subcritical by inserting all control banks into the core. The shutdown rods may remain withdrawn.

c. When the average reactor coolant loop temperature is greater than 350°F, the following conditions shall be met:

1. At least two reactor coolant loops shall be OPERABLE.

2. At least one reactor coolant loop shall be in operation.

d. When the average reactor coolant loop temperature is less than or equal to 350°F, the following conditions shall be met:

1. A minimum of two non-isolated loops, consisting of any combination of reactor coolant loops or residual heat removal loops, shall be OPERABLE, except as specified below:

    (a) One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

    (b) During REFUELING OPERATIONS the residual heat removal loop may be removed from operation as specified in TS 3.10.A.6.

2. At least one reactor coolant loop or one residual heat removal loop shall be in operation, except as specified in Specification 3.10.A.6.

e. When all three pumps have been idle for > 15 minutes, the first pump shall not be started unless: (1) a bubble exists in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

2. Steam Generator

A minimum of two steam generators in non-isolated loops shall be OPERABLE when the average Reactor Coolant System temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. Three valves shall be OPERABLE when the head is on the reactor vessel and the Reactor Coolant System average temperature is greater than 350°F, the reactor is critical, or the Reactor Coolant System is not connected to the Residual Heat Removal System.

b. Valve lift settings shall be maintained at 2485 psig ± 1 percent*

* The as-found tolerance shall be ±3% and the as-left tolerance shall be ±1%.

4. Reactor Coolant Loops

a. Loop stop valves shall not be closed in more than one loop unless the Reactor Coolant System is connected to the Residual Heat Removal System and the Residual Heat Removal System is OPERABLE.

b. POWER OPERATION with less than three loops in service is prohibited. The following loop isolation valves shall have AC power removed and their breakers locked, sealed or otherwise secured in the open position during POWER OPERATION:

Unit No. 1	Unit No. 2
MOV 1590	MOV 2590
MOV 1591	MOV 2591
MOV 1592	MOV 2592
MOV 1593	MOV 2593
MOV 1594	MOV 2594
MOV 1595	MOV 2595

5. Pressurizer

a. The reactor shall be maintained subcritical by at least 1% until the steam bubble is established and the necessary sprays and at least 125 KW of heaters are operable.

b. With the pressurizer inoperable due to inoperable pressurizer heaters, restore the inoperable heaters within 72 hours or be in at least HOT SHUTDOWN within 6 hours and the Reactor Coolant System temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours.

c. With the pressurizer otherwise inoperable, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and the Reactor Coolant System temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours.

6. Relief Valves

Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE* whenever the Reactor Coolant System average temperature is ≥350°F.

a. With one or both PORVs inoperable but capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain power to the associated block valve(s). Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and reduce Reactor Coolant System average temperature to <350°F within the following 6 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to OPERABLE status or capable of being manually cycled or close the associated block valve and remove power from the block valve. In addition, restore the PORV to OPERABLE status or capable of being manually cycled within the following 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and reduce Reactor Coolant System average temperature to <350°F within the following 6 hours.

c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour restore at least 1 PORV to OPERABLE status or capable of being manually cycled. Otherwise, close the associated block valves and remove power from the block valves. In addition, be in HOT SHUTDOWN within the next 6 hours and reduce Reactor Coolant System average temperature to <350°F within the following 6 hours.

*Automatic actuation capability may be blocked when Reactor Coolant System pressure is below 2000 psig.

d. With one block valve inoperable, within 1 hour either restore the block valve to OPERABLE status or place the associated PORV in manual In addition, restore the block valve to OPERABLE status in the next 72 hours or, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant average temperature to <350°F within the following 6 hours.

e. With both block valves inoperable, within 1 hour either restore the block valves to OPERABLE status or place the associated PORVs in manual. Restore at least 1 block valve to OPERABLE status within the next hour or, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant average temperature to <350°F within the following 6 hours

f With one or both PORV(s) inoperable (but capable of being manually cycled) because of an inoperable backup air supply, within 14 days either restore the PORV(s) backup air supply(ies) to OPERABLE status or be in at least HOT SHUTDOWN within the next 6 hours and reduce Reactor Coolant System average temperature to < 350°F within the following 6 hours.

7. Reactor Vessel Head Vents

a. At least two Reactor Vessel Head vent paths consisting of two isolation valves in series powered from emergency buses shall be OPERABLE and closed whenever RCS temperature and pressure are >350°F and 450 psig.

b. With one Reactor Vessel Head vent path inoperable; startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of both isolation valves in the inoperable vent path.

c. With two Reactor Vessel Head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuator of all isolation valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 30 days or be in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

Basis

Specification 3.1.A-1 requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling flow in the event of a loss of reactor coolant flow accident. This provided flow will maintain the DNBR above the applicable design limit.(1) Heat transfer analyses also show that reactor heat equivalent to approximately 10% of rated power can be removed with natural circulation; however, the plant is not designed for critical operation with natural circulation or one loop operation and will not be operated under these conditions.

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 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 reactor coolant system volume in approximately one half hour.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The requirement for redundant coolant loops ensures the capability to remove core decay heat when the Reactor Coolant System average temperature is less than or equal to 350°F. Because of the low-low steam generator water level reactor trip, normal reactor criticality cannot be achieved without water in the steam generators in reactor coolant loops with open loop stop valves. The requirement for two OPERABLE steam generators, combined with the requirements of Specification 3.6, ensure adequate heat removal capabilities for Reactor Coolant System temperatures of greater than 350°F.

Each of the pressurizer safety valves is designed to relieve 295,000 lbs. per hr. of saturated steam at the valve setpoint. Two safety valves have a capacity greater than the maximum surge rate resulting from complete loss of load.(2)

The limitation specified in item 4 above on reactor coolant loop isolation will prevent an accidental isolation of all the loops which would eliminate the capability of dissipating core decay heat when the Reactor Coolant System is not connected to the Residual Heat Removal System.

The requirement for steam bubble formation in the pressurizer when the reactor passes 1% subcriticality will ensure that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that 125 Kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT SHUTDOWN.

The power operated relief valves (PORVs) operate to relieve Reactor Coolant System pressure below the setting of the pressurizer code safety valves. The PORVs and their associated block valves may be used by the unit operators to depressurize the Reactor Coolant System to recover from certain transients if normal pressurizer spray is not available. Specifically, cycling of the PORVs is required to mitigate the consequences of a design basis steam generator tube rupture accident. Therefore, whenever a PORV is inoperable, but capable of being manually cycled, the associated block valve will be closed with its power maintained. The capability to cycle the PORVs is verified during each refueling outage (and is not required during power operations). These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve leak excessively. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible Reactor Coolant System leakage path.

With one or both PORVs inoperable (but capable of being manually cycled) due to an inoperable backup air supply, continued operation for 14 days is allowed provided the normal motive force for the PORVs, i.e., the instrument air system, continues to be available. Instrument air has a high system reliability, and the likelihood of it being unavailable during a demand for PORV operation is low enough to justify a reasonable length of time (i.e , 14 days) to repair the backup air system

The accumulation of non-condensable gases in the Reactor Coolant System may result from sudden depressurization, accumulator discharges and/or inadequate core cooling conditions. The function of the Reactor Vessel Head Vent is to remove non-condensable gases from the reactor vessel head. The Reactor Vessel Head Vent is designed with redundant safety grade vent paths. Venting of non-condensable gases from the pressurizer steam space is provided primarily through the Pressurizer PORVs. The pressurizer is, however, equipped with a steam space vent designed with redundant safety grade vent paths

<u>References</u>

(1) UFSAR Section 14.2 9

(2) UFSAR Section 14.2.10

B. HEATUP AND COOLDOWN

Specification

1. Unit 1 and Unit 2 reactor coolant temperature and pressure and the system heatup and cooldown (with the exception of the pressurizer) shall be limited in accordance with TS Figures 3.1-1 and 3.1-2.

   Heatup:

   Figure 3.1-1 may be used for heatup rates of up to 60°F/hr.

   Cooldown:

   Allowable combinations of pressure and temperature for specific cooldown rates are below and to the right of the limit lines as shown in TS Figure 3.1-2. This rate shall not exceed 100°F/hr. Cooldown rates between those shown can be obtained by interpolation between the curves on Figure 3.1-2.

   Core Operation:

   During operation where the reactor core is in a critical condition (except for low level physics tests), vessel metal and fluid temperature shall be maintained above the reactor core criticality limits specified in 10 CFR 50 Appendix G. The reactor shall not be made critical when the reactor coolant temperature is below 522°F as specified in T.S. 3.1.E.

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

3. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. and 200°F/hr., respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2.

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

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

2) These limit lines shall be calculated periodically using methods provided below.

3) 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.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. and 200°F/hr. respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI according to the leak test limit line shown in Figure 3.1-1.

6) The reactor shall not be made critical when the reactor coolant temperature is below 522°F in accordance with Technical Specification 3.1.E.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 28.8 Effective Full Power Years (EFPY) and 29.4 EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (228.4°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 28.8 EFPY or 29.4 EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

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

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

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was 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. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

References

(1) UFSAR, Section 4.1, Design Bases

C. Leakage

Specifications

1. Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment.
2. 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 four hours of leak detection, the reactor shall be brought to hot shutdown. If the source of leakage is not identified within an additional 48 hours, the reactor shall be brought to a cold shutdown condition.
3. 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 C.4 below.
4. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System component body, pipe well, vessel wall, or pipe weld, the reactor shall be brought to a cold shutdown condition and corrective action taken prior to resumption of unit operation.
5. If the total leakage,other than leakage from controlled sources, exceeds 10 gpm the reactor shall be placed in the cold shutdown condition.

6. If the primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System exceeds 1 gpm total and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System, reduce the leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

7. a. Prior to going critical all primary coolant system pressure isolation valves listed below shall be functional as a pressure isolation device, except as specified in 3.1.C.7.b. Valve leakage shall not exceed the amounts indicated.

	Unit 1	Unit 2	Max. Allowable Leakage (see note (a) below)
Loop A, Cold Leg	1-SI-79, 1-SI-241	2-SI-79, 2-SI-241	≤5.0 gpm for each valve
Loop B, Cold Leg	1-SI-82, 1-SI-242	2-SI-82, 2-SI-242	
Loop C, Cold Leg	1-SI-85, 1-SI-243	2-SI-85, 2-SI-243	

b. If Specification 3.1.C.7.a cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown within 6 hours and in the cold shutdown condition within the following 30 hours.

Notes

(a). 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

Basis

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, the Gaseous and Liquid Waste Disposal Systems, the Component Cooling System, and the Chemical and Volume Control System.

Detection of leaks from the Reactor Coolant System is by one or more of the following:

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 condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor.

References

FSAR, Section 4.2.7 - Reactor Coolant System Leakage

FSAR, Section 14.3.2 - Rupture of a Main Steam Pipe

D. Maximum Reactor Coolant Activity

Specifications

1. The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed $100/\bar{E}$ µCi/cc whenever the reactor is critical or the average temperature is greater than 500°F, where $\bar{E}$ is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection.

2. The specific activity of the reactor coolant shall be limited to $\leq$ 1.0 µCi/cc DOSE EQUIVALENT I-131 whenever the reactor is critical or the average temperature is greater than 500°F.

3. The requirements of D-2 above may be modified to allow the specific activity of the reactor coolant > 1.0 µCi/cc DOSE EQUIVALENT I-131 but less than 10.0 µCi/cc DOSE EQUIVALENT I-131. Following shutdown, the unit may be restarted and/or operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. With the specific activity of the reactor coolant > 1.0 µCi/cc DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10.0 µCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.

4. If the specific activity of the reactor coolant exceeds 1.0 µCi/cc DOSE EQUIVALENT I-131 or $100/\overline{E}$ µCi/cc, a report shall be prepared and submitted to the Commission pursuant to Specification 6.6.A.2. This report shall contain the results of the specific activity analysis together with the following information:

   a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

   b. Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded,

c. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded,

d. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/cc DOSE EQUIVALENT I-131,

e. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded, and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations, and

f. Graph of the I-131 concentration and one other radioiodine isotope concentration in μCi/cc as a function of time for the duration of the specific activity above the steady-state level.

Basis

The specified limit provides protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the following analysis of a steam generator tube rupture accident in UFSAR Chapter 14.3.1.

Rupture of a steam generator tube would allow radionuclides in the reactor coolant to enter the secondary system. The limiting case involves a double-ended tube rupture coincident with loss of the condenser and release of steam from the secondary side to the atmosphere via the main steam safety valves or atmospheric relief valves. This is assumed to continue for 30 minutes in the analysis. The operator will take action to reduce the primary side temperature to a value below that corresponding to the relief or safety valve setpoint. Once this is accomplished the valves can be closed and the release terminated.

Permitting startup and/or REACTOR OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 µCi/cc but < 10.0 µCi/cc DOSE EQUIVALENT I-131 accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER. Although the analysis of a steam generator tube rupture initiated with primary coolant activity at the 10.0 µCi/cc transient limit shows offsite doses well within the 10 CFR 100 limits, operation at the transient limit is restricted to no more than 10 percent of the unit's yearly operating time to limit the risk of appreciable releases following a postulated steam generator tube rupture.

The basis for the 500°F temperature contained in the Specification is that the saturation pressure corresponding to 500°F, i.e., 680.8 psia, is well below the pressure at which the atmospheric relief valves on the secondary side could be actuated.

The accident analysis examines two cases of iodine spiking. For the case with a pre-existing iodine spike, the transient coolant activity limit of 10.0 µCi/cc is assumed. For the case of a concurrent spike, the initial activity is assumed to correspond to the steady state limit of 1.0 µCi/cc. The concurrent iodine spike is modeled with a conservative iodine appearance rate. Both cases show doses at the exclusion area and low population zone boundaries which are well within the 10 CFR Part 100 limits and control room doses which are within the General Design Criterion (GDC) 19 guidelines.

Measurement of $\bar{E}$ will be performed at least twice annually. Calculations required to determine $\bar{E}$ will consist of the following:

1. $\bar{E}$ shall be the average (weighed in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.
2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes.
3. A calculation of $\bar{E}$ by appropriate weighing of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/cc) 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 either: a) those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", or b) Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR Part 50, Appendix I."

E. Minimum Temperature for Criticality

Specifications

1. Except during LOW POWER PHYSICS TESTS, the reactor shall not be made critical at any Reactor Coolant System temperature above which the moderator temperature coefficient is more positive than the limit specified in the CORE OPERATING LIMITS REPORT. The maximum upper limit for the moderator temperature coefficient shall be:

   a. + 6 pcm/°F at less than 50% of RATED POWER, or
   b. + 6 pcm/°F at 50% of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER.

2. In no case shall the reactor be made critical with the Reactor Coolant System temperature below the limiting value of RT_{NDT} + 10°F, where the limiting value of RT_{NDT} is as determined in Part B of this specification.

3. When the Reactor Coolant System temperature is below the minimum temperature as specified in E-2 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization.

4. The reactor shall not be made critical when the Reactor Coolant System temperature is below 522°F.

Basis

During the early part of a fuel cycle, the moderator temperature coefficient may be calculated to be slightly positive at coolant temperatures in the power operating range. The moderator coefficient will be most positive at the beginning of cycle life, when the boron concentration in the coolant is the greatest. Later in the cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be less positive or will be negative in the power operating range. At the beginning of cycle life, during pre-operational physics tests, measurements are made to determine that the moderator coefficient is less than the limit specified in the CORE OPERATING LIMITS REPORT.

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the low power limit specified in the CORE OPERATING LIMITS REPORT has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during LOW POWER PHYSICS TESTS to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operation precautions will be taken. In addition, the strong negative Doppler coefficient (2)(3) and the small integrated Delta k/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below the limiting value of RT_{NDT} + 10°F provides increased assurance that the proper relationship between Reactor Coolant System pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility transition temperature range. Heatup to this temperature is accomplished by operating the reactor coolant pumps.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below 522°F provides added assurance that the assumptions made in the safety analyses remain bounding by maintaining the moderator temperature within the range of those analyses.

If a specified shutdown reactivity margin is maintained (TS Section 3.12), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.

(1) UFSAR Figure 3.3-8
(2) UFSAR Table 3.3-1
(3) UFSAR Figure 3.3-9

F. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

Specification

1. Concentrations of contaminants in the reactor shall not exceed any one of the following limits when the reactor coolant is above 250°F.

Contaminant	Normal Steady-State Operation (PPM)	Transients not to Exceed 24 Hours (PPM)
a. Oxygen	0.10	1.00
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

2. If any one of the normal steady-state operating limits as specified in 3.1.F.1 above are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any one of the contaminants can not be controlled within the limits of Specification 3.1.F.1 above, the reactor shall be brought to the cold shutdown condition, utilizing normal operating procedures, and the cause of the out-of-specification operation ascertained and corrected. The reactor may then be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise, a safety review is required before startup.

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

Contaminant	Normal Concentration (PPM)	Transients not to exceed 24 hours (PPM)
a. Chloride	0.15	1.5
b. Fluoride	0.15	1.5

If the limits above are exceeded, the reactor shall be immediately brought to COLD SHUTDOWN and the cause of the out-of-specification condition shall be ascertained and corrected.

5. For the purposes of correcting the contaminant concentrations to meet Technical Specifications 3.1.F.1 and 3.1.F.4 above, increase in coolant temperature consistent with operation of primary coolant pumps for a short period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250 degrees F.

6. For conditions above COLD SHUTDOWN, if more than one contaminant or contaminants transient, which results in contaminant levels exceeding any of the normal steady state operation limits specified in 3.1.F.1 or 3.1.F.4, is experienced in any seven consecutive day period, the reactor shall be placed in COLD SHUTDOWN until the cause of the out-of-specification operation is ascertained and corrected.

Basis

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in technical specification 3.1.F.1 and 3.1.F.4 the integrity of the reactor coolant system is assured under all operating conditions.[(1)] If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin, or adjustment of the hydrogen concentration in the volume control tank.[(2)] Because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is not necessary to shutdown immediately if the condition can be corrected. Thus the period of 24 hours for corrective action to restore concentrations within the 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 COLD SHUTDOWN and the corrective action will continue.

In restoring 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 which will not increase the average coolant temperature above 250°F.

More than one contaminant transient, in any seven consecutive day period, that results in exceeding normal steady state operation limits, could be indicative of unforeseen chemistry control problems. Such potential problems warrant investigation, correction and measures to insure that the integrity of the Reactor Coolant System is maintained.

References

(1) UFSAR 4.2
(2) UFSAR 9.2

G. Reactor Coolant System Overpressure Mitigation

Specification

1. The Reactor Coolant System (RCS) overpressure mitigating system shall be OPERABLE as described below:

   a. Whenever the RCS average temperature is greater than 350°F, a bubble shall exist in the pressurizer with the necessary sprays and heaters OPERABLE.

   b. Prior to decreasing RCS average temperature below 350°F, verify a maximum of one charging pump is capable of injecting into the RCS and that each accumulator is isolated. Thereafter, once per 12 hours:

      (1) Verify that a maximum of one charging pump is capable of injecting into the RCS.

      (2) Verify that each accumulator is isolated, if isolation is required.

   c. Whenever the RCS average temperature is less than or equal to 350°F and the reactor vessel head is bolted:

      (1) A maximum of one charging pump shall be OPERABLE and capable of injecting into the RCS. Two charging pumps may be in operation momentarily during transfer of operation from one charging pump to another.

      and

      (2) The accumulators shall be isolated (accumulator discharge valves closed and their respective breakers locked, sealed or otherwise secured in the open position). Isolation is not required if the accumulator pressure is less than the pressurizer PORV setpoint specified in TS 3.1.G.1.c.(4).

      and

(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of ≤ 390 psig and verify each PORV block valve is open at least once per 72 hours,

or

(5) The RCS shall be vented through one open PORV or an equivalent size opening as specified below:

(a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or

(b) with the RCS vented through a locked open vent path verify the path is open at least once per 31 days.

2. The requirements of Specification 3.1.G.1.c.(4) may be modified as follows:

a One PORV may be inoperable in INTERMEDIATE SHUTDOWN with the RCS average temperature > 200°F but < 350°F for a period not to exceed 7 days. If the inoperable PORV is not restored to OPERABLE status within 7 days, then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within the next 8 hours.

b One PORV may be inoperable in COLD SHUTDOWN or REFUELING SHUTDOWN with the reactor vessel head bolted for a period not to exceed 24 hours. If the inoperable PORV is not restored to OPERABLE status within 24 hours then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within 8 hours.

c. With both PORV's inoperable, depressurize the RCS within 8 hours unless Specification 3.1.G.1.c.(3) is in effect. When the RCS has been depressurized, vent the RCS through one open PORV or an equivalent sized opening, or establish the conditions listed below. Maintain the RCS depressurized until both PORV's have been restored to OPERABLE status.

(1) A maximum pressurizer narrow range level of 33%.
(2) The series RHR inlet valves open and their respective breakers locked open or an alternate letdown path OPERABLE.
(3) A maximum of one charging pump is capable of injecting into the RCS.
(4) Safety Injection accumulator discharge valves closed and their respective breakers locked, sealed, or otherwise secured in the open position.

d. When the conditions noted in 3.1.G.2.c.(1) through 3.1.G.2.c.(4) above are required to be established, verify the required conditions are met at least once per 12 hours.

3. In the event that the Reactor Coolant System Overpressure Mitigating System is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the mitigating system or the administrative controls on the transient and any corrective actions necessary to prevent recurrence.

<u>Basis</u>

The operability of two PORV's or the RCS vented through an opened PORV ensures that the Reactor Vessel will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the Reactor Coolant System average temperature is $\leq$ 350°F and the Reactor Vessel Head is bolted. When the Reactor Coolant System average temperature is > 350°F, overpressure protection is provided by a bubble in the pressurizer and/or pressurizer safety valves. A single PORV has adequate relieving capability to protect the Reactor Vessel from overpressurization when the transient is limited to either (1) the start of an idle Reactor Coolant Pump with the secondary water temperature of a steam generator ≤ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

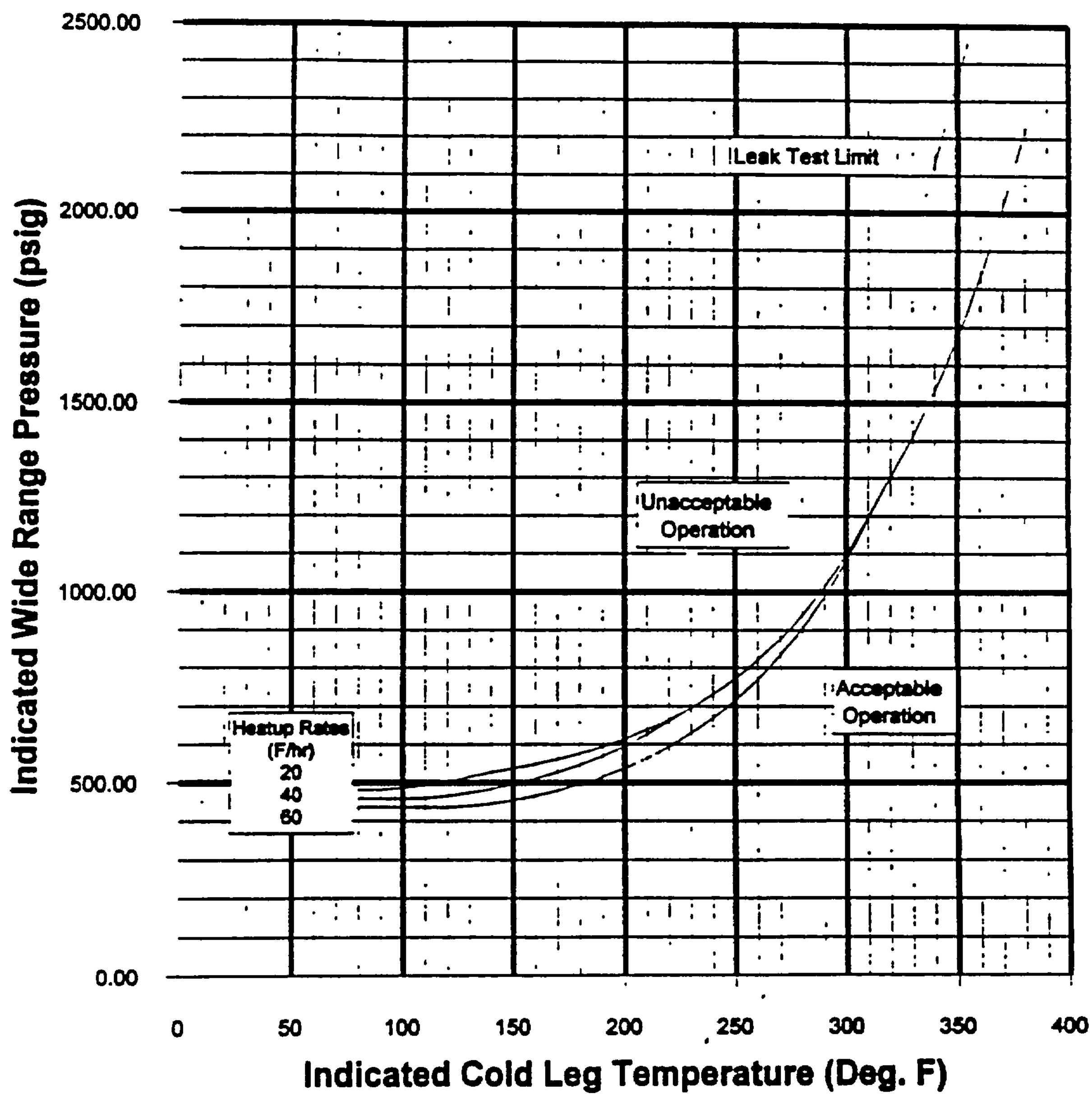
The limitation for a maximum of one charging pump allowed OPERABLE and the surveillance required to verify that two charging pumps are inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV, or equivalent. The Safety Injection accumulators are not considered a credible mass input mechanism for RCS low temperature overpressurization concerns. There are administrative controls to ensure isolation, including de-energizing the Safety Injection (SI) accumulator isolation valves, during plant shutdown conditions (RCS pressure less than 1000 psig) to prevent inadvertent SI accumulator discharge into the RCS for low temperature overpressurization concerns. An undesired pressurizer PORV lift due to inadvertent SI accumulator discharge is not possible when SI accumulator pressure is less than the low temperature PORV lift setpoint specified in TS 3.1.G. Therefore, SI accumulator isolation, and verification of such isolation is not necessary when SI accumulator pressure is less than the low temperature PORV setpoint.

A maximum pressurizer narrow range level of 33% has been selected to provide sufficient time, approximately 10 minutes, for operator response in case of a malfunction resulting in maximum charging flow from one charging pump (530 gpm). Operator action would be initiated by at least two alarms that would occur between the normal operating level and the maximum allowable level (33%). When both PORVs are inoperable and it is impossible to manually open at least one PORV, additional administrative controls shall be implemented to prevent a pressure transient that would exceed the limits of Appendix G to 10 CFR Part 50.

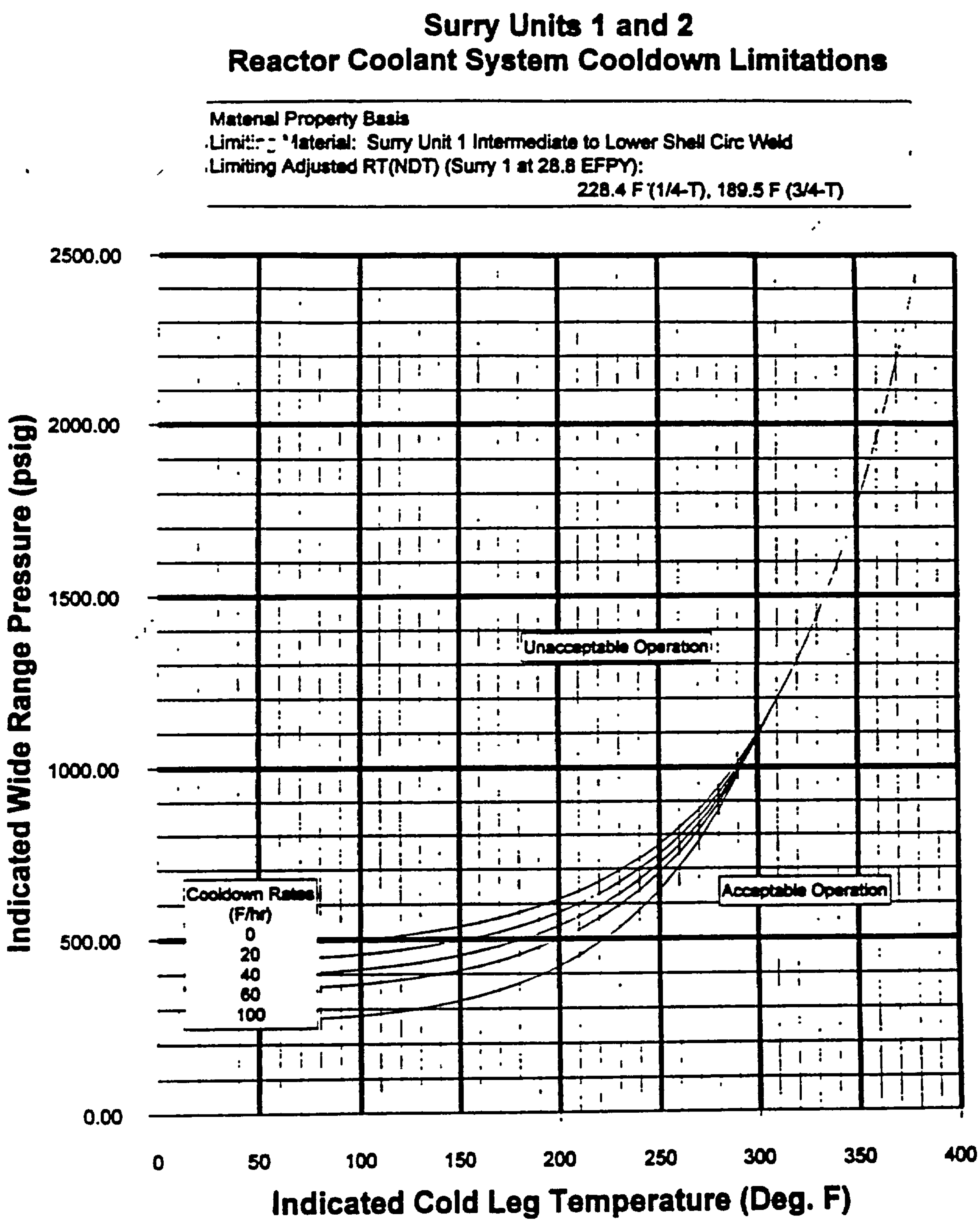
The requirements of this specification are only applicable when the Reactor Vessel head is bolted. When the Reactor Vessel head is unbolted, a RCS pressure of < 100 psig will lift the head, thereby creating a relieving capability equivalent to at least one PORV.

Surry Units 1 and 2
Reactor Coolant System Heatup Limitations

Material Property Basis
Limiting Material: Surry Unit 1 Intermediate to Lower Shell Circ Weld
Limiting Adjusted RT(NDT) (Surry 1 at 28.8 EFPY):
228.4F (1/4-T), 189.5 F (3/4-T)



Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2



Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

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 ensure safe reactor operation.

Specification

A. When fuel is in a reactor, there shall be at least one flow path to the core for boric acid injection. The minimum capability for boric acid injection shall be equivalent to that supplied from the refueling water storage tank.

B. The reactor shall not be critical unless:
   1. At least two boron injection subsystems are OPERABLE consisting of:
      a. A Chemical and Volume Control subsystem consisting of:
         1. One OPERABLE flow path,
         2. One OPERABLE charging pump,
         3. One OPERABLE boric acid transfer pump,
         4. The common OPERABLE boric acid storage system with:
            a. A minimum contained borated water volume of 6000 gallons per unit,
            b. A boron concentration of at least 7.0 weight percent but not more than 8.5 weight percent boric acid solution, and
            c. A minimum solution temperature of 112°F.
            d. An OPERABLE boric acid transfer pump for recirculation.

b. A subsystem supplying borated water from the refueling water storage tank via a charging pump to the Reactor Coolant System consisting of:

1. One OPERABLE flow path,
2. One OPERABLE charging pump,
3. The OPERABLE refueling water storage tank with:
   a. A minimum contained borated water volume of 387,100 gallons,
   b. A boron concentration of at least 2300 ppm but not more than 2500 ppm, and
   c. A maximum solution temperature of 45°F.

2. One charging pump from the opposite unit is available with:

   a. the pump being OPERABLE except for automatic initiation instrumentation,
   b. offsite or emergency power may be inoperable when in COLD SHUTDOWN, and
   c. the pump capable of being used for alternate shutdown with the opening of the charging pump cross-connect valves.

C. The requirements of Specification 3.2.B may be modified as follows:

1. With only one of the boron injection subsystems OPERABLE, restore at least two boron injection subsystems to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours.
2. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or place the reactor in HOT SHUTDOWN within the next 6 hours.

   a. For conditions where the RWST is inoperable due to boron concentration or solution temperature not being within the limits of Specification 3.3.A.1, restore the parameters to within specified limits in 8 hours or place the reactor in HOT SHUTDOWN within the next 6 hours.

3. With no charging pump from the opposite unit available, return at least one of the opposite unit's charging pumps to available status in accordance with Specification 3.2.B.2 within 7 days or place the reactor in HOT SHUTDOWN within the next 6 hours.

D. If the requirements of Specification 3.2.B are not satisfied as allowed by Specification 3.2.C, the reactor shall be placed in COLD SHUTDOWN within the following 30 hours.

E. During REFUELING SHUTDOWN and COLD SHUTDOWN the following valves in the affected unit shall be locked, sealed, or otherwise secured in the closed position except during planned dilution or makeup activities:

1. During Unit 1 REFUELING SHUTDOWN and COLD SHUTDOWN:
   a. Valve 1-CH-223, or
   b. Valves 1-CH-212, 1-CH-215, and 1-CH-218.

2. During Unit 2 REFUELING SHUTDOWN and COLD SHUTDOWN:
   a. Valve 2-CH-223, or
   b. Valves 2-CH-212, 2-CH-215, and 2-CH-218.

3. Following a planned dilution or makeup activities, the valves listed in Specifications 3.2.E.1 and 3.2.E.2 above, for the affected unit, shall be locked, sealed, or otherwise secured in the closed position within 15 minutes.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using boric acid transfer pumps which discharge to the suction of each unit's charging pumps. The Chemical and Volume Control System contains four boric acid transfer pumps. Two of these pumps are normally assigned to each unit but, valving and piping arrangements allow pumps to be shared such that three out of four pumps can service either unit. An alternate (not normally used) method of boration is to use the charging pumps taking suction directly from the refueling water storage tank. There are two sources of borated water available to the suction of the charging pumps through two different paths; one from the refueling water storage tank and one from the discharge of the boric acid transfer pumps.

A. The boric acid transfer pumps can deliver the boric acid tank contents (7.0% solution of boric acid) to the charging pumps.

B. The charging pumps can take suction from the volume control tank, the boric acid transfer pumps and the refueling water storage tank. Reference is made to Technical Specification 3.3.

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 6000 gallons of the 7.0% solution of boric acid are required to meet COLD SHUTDOWN conditions. Thus, a minimum of 6000 gallons in the boric acid tank is specified. An upper concentration limit of 8.5% boric acid in the tank is specified to maintain solution solubility at the specified low temperature limit of 112 degrees F.

The Boric Acid Tank(s) are supplied with level alarms which would annunciate if a leak in the system occurred.

For one-unit operation, it is required to maintain available one charging pump with a source of borated water on the opposite unit, the associated piping and valving, and the associated instrumentation and controls in order to maintain the capability to cross-connect the two unit's charging pump discharge headers. In the event the operating unit's charging pumps become inoperable, this permits the opposite unit's charging pump to be used to bring the disabled unit to COLD SHUTDOWN conditions. Initially, the need for the charging pump cross-connect was identified during fire protection reviews.

The requirement that certain valves remain closed during REFUELING SHUTDOWN and COLD SHUTDOWN conditions, except for planned boron dilution or makeup activities, provides assurance that an inadvertent boron dilution will not occur. This specification is not applicable at INTERMEDIATE SHUTDOWN, HOT SHUTDOWN, REACTOR CRITICAL, or POWER OPERATION.

<u>References</u>

UFSAR Sections 9.1 Chemical and Volume Control System

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless:

1. The refueling water storage tank (RWST) is OPERABLE with:

    a. A contained borated water volume of at least 387,100 gallons.

    b. A boron concentration of at least 2300 ppm but not greater than 2500 ppm.

    c. A maximum solution temperature of 45° F.

2. Each safety injection accumulator is OPERABLE with:

    a. A borated water volume of at least 975 cubic feet but not greater than 1025 cubic feet.

    b. A boron concentration of at least 2250 ppm.

    c. A nitrogen cover-pressure of at least 600 psia.

    d. The safety injection accumulator discharge motor operated valve blocked open by de-energizing AC power and the valves's breaker locked, sealed or otherwise secured in the open position when the reactor coolant system pressure is greater than 1000 psig.

3. Two safety injection subsystems are OPERABLE with subsystems comprised of:

   a. One OPERABLE high head charging pump.
   b. One OPERABLE low head safety injection pump.
   c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation.

B. The requirements of Specification 3.3.A may be modified as follows:

1. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or place the reactor in HOT SHUTDOWN within the next 6 hours.

   a. For conditions where the RWST is inoperable due to boron concentration or solution temperature not being within the limits of Specification 3.3.A.1, restore the parameters to within specified limits in 8 hours or place the reactor in HOT SHUTDOWN within the next 6 hours.

2. With one safety injection accumulator inoperable, restore the accumulator to OPERABLE status within 4 hours or place the reactor in HOT SHUTDOWN within the next 6 hours.

   a. For conditions where one safety injection accumulator is inoperable due to boron concentration not being within the limits of Specification 3.3.A.2, restore the accumulator to within specified limits in 72 hours or place the reactor in HOT SHUTDOWN within the next 6 hours.
   b. Power may be restored to any valve or breaker referenced in Specification 3.3.A.2.d for the purpose of testing or

maintenance provided that not more than one valve has power restored, and the testing and maintenance is completed and power removed within 4 hours.

3. With one safety injection subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or place the reactor in HOT SHUTDOWN within the next 6 hours.

C. If the requirements of Specification 3.3.A are not satisfied as allowed by Specification 3.3.B, the reactor shall be placed in COLD SHUTDOWN in the following 30 hours.

Basis

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. With this mode of startup the Safety Injection System is required to be OPERABLE as specified. During LOW POWER PHYSICS TESTS there is a negligible amount of energy stored in the system. Therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the full capacity of the Safety Injection System would not be necessary.

The OPERABLE status of the subsystems is to be demonstrated by periodic tests, detailed in TS Section 4.11. A large fraction of these tests are performed while the reactor is operating in the power range. If a subsystem is found to be inoperable, it will be possible in most cases to effect repairs and restore the subsystem to full operability within a relatively short time. A subsystem being inoperable 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 subsystem failures. In some cases, additional components (i.e., charging pumps) are installed to allow a component to be inoperable without affecting system redundancy.

If the inoperable subsystem is not repaired within the specified allowable time period, the reactor will initially be placed in HOT SHUTDOWN to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. If the malfunction(s) is not corrected the reactor will be placed in COLD SHUTDOWN following normal shutdown and cooldown procedures.

Assuming the reactor has been operating at full RATED POWER for at least 100 days, the magnitude of the decay heat production decreases as follows after a unit trip from full RATED POWER.

Time After Shutdown	Decay Heat, (% of RATED POWER)
1 min.	3.7
30 min.	1.6
1 hour	1.3
8 hours	0.75
48 hours	0.48

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident, while in HOT SHUTDOWN, is reduced by orders of magnitude below the requirements for handling a postulated loss-of-coolant accident occurring during POWER OPERATION. Placing and maintaining the reactor in HOT SHUTDOWN significantly reduces the potential consequences of a loss-of-coolant accident, allows access to some of the Safety Injection System components in order to effect repairs, and minimizes the plant's exposure to thermal cycling.

Failure to complete repairs within 72 hours is considered indicative of unforeseen problems (i.e., possibly the need of major maintenance). In such a case, the reactor is placed in COLD SHUTDOWN.

The accumulators are able to accept leakage from the Reactor Coolant System without any effect on their operability. Allowable inleakage is based on the volume of water that can be added to the initial amount without exceeding the volume given in Specification 3.3.A.2.

The accumulators (one for each loop) discharge into the cold leg of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motor-operated valve to isolate the accumulator during reactor start-up and shutdown to preclude the discharge of the contents of the accumulator when not required.

Accumulator Motor Operated Discharge Isolation Valves

Unit No. 1	Unit No. 2
MOV 1865A	MOV 2865A
MOV 1865B	MOV 2865B
MOV 1865C	MOV 2865C

However, to assure that the accumulator valves satisfy the single failure criteria, they will be locked, sealed or otherwise secured open by de-energizing the valve motor operators when the reactor coolant pressure exceeds 1000 psig. The operating pressure of the Reactor Coolant System is 2235 psig and accumulator injection is initiated when this pressure drops to 600 psia. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal startup operation to perform the actions required to de-energize the valve. This procedure will assure that there is an OPERABLE flow path from each accumulator to the Reactor Coolant System during POWER OPERATION and that safety injection can be accomplished.

The removal of power from the valves listed above will assure that the systems of which they are a part satisfy the single failure criterion.

3.4 SPRAY SYSTEMS

Applicability

Applies to the operational status of the Spray Systems.

Objective

To define those limiting conditions for operation of the Spray Systems necessary to assure safe unit operation.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not be made to exceed 350°F or 450 psig, respectively, unless the following Spray System conditions in the unit are met:

1. Two Containment Spray Subsystems, including containment spray pumps, piping, and valves shall be OPERABLE.

2. Four Recirculation Spray Subsystems, including recirculation spray pumps, coolers, piping, and valves shall be OPERABLE.

3. The refueling water storage tank shall contain at least 387,100 gallons of borated water at a maximum temperature of 45°F. The boron concentration shall be at least 2300 ppm but not greater than 2500 ppm.

4. The refueling water chemical addition tank shall contain at least 3930 gallons of solution with a sodium hydroxide concentration of at least 17 percent by weight but not greater than 18 percent by weight.

5. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions shall be OPERABLE.

B. During POWER OPERATION the requirements of Specification 3.4.A may be modified to allow a subsystem or the following components to be inoperable. If the components are not restored to meet the requirements of Specification 3.4.A within the time period specified below, the reactor shall be placed in HOT SHUTDOWN within the next 6 hours. If the requirements of Specification 3.4.A are not satisfied within an additional 48 hours the reactor shall be placed in COLD SHUTDOWN within the following 30 hours.

1. One Containment Spray Subsystem may be inoperable, provided immediate attention is directed to making repairs and the subsystem can be restored to OPERABLE status within 24 hours.

2. One outside Recirculation Spray Subsystem may be inoperable, provided immediate attention is directed to making repairs and the subsystem can be restored to OPERABLE status within 24 hours.

3. One inside Recirculation Spray Subsystem may be inoperable, provided immediate attention is directed to making repairs and the subsystem can be restored to OPERABLE status within 72 hours.

4. Refueling Water Storage Tank volume may be outside the limits of Specification 3.4.A.3 provided it is restored to within limits within one hour.

   a. For conditions where the RWST is inoperable due to boron concentration or solution temperature not being within the limits specified, restore the parameters to within specified limits in 8 hours.

Basis

The spray systems in each reactor unit consist of two separate parallel Containment Spray Subsystems, each of 100 percent capacity, and four separate parallel Recirculation Spray Subsystems, each of 50 percent capacity.

Each Containment Spray Subsystem draws water independently from the refueling water storage tank (RWST). The water in the tank is cooled to 45°F or below by circulating the water through one of the two RWST coolers with one of the two recirculating pumps. The water temperature is maintained by two mechanical refrigerating units as required. In each Containment Spray Subsystem, the water flows from the tank through an electric motor driven containment spray pump and is sprayed into the containment atmosphere through two separate sets of spray nozzles. The capacity of the spray systems to depressurize the containment in the event of a Design Basis Accident is a function of the pressure and temperature of the containment atmosphere, the service water temperature, and the temperature in the refueling water storage tank as discussed in the Basis of Specification 3.8.

Each Recirculation Spray Subsystem draws water from the common containment sump. In each subsystem the water flows through a recirculation spray pump and recirculation spray cooler, and is sprayed into the containment atmosphere through a separate set of spray nozzles. Two of the recirculation spray pumps are located inside the containment and two outside the containment in the containment auxiliary structure.

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the spray systems are capable of cooling and depressurizing the containment to 0.5 psig in less than 60 minutes and to subatmospheric pressure within 4 hours following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air inleakage. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (from 1 hour to 4 hours) and is maintained less than 0.0 psig (after 4 hours).

In addition to supplying water to the Containment Spray System, the refueling water storage tank is also a source of water for safety injection following an accident. This water is borated to a concentration which assures reactor shutdown by approximately 5 percent $\Delta k/k$ when all control rods assemblies are inserted and when the reactor is cooled down for refueling.

References

UFSAR Section 4	Reactor Coolant System
UFSAR Section 6.3.1	Containment Spray Subsystem
UFSAR Section 6.3.1	Recirculation Spray Pumps and Coolers
UFSAR Section 6.3.1	Refueling Water Chemical Addition Tank
UFSAR Section 6.3.1	Refueling Water Storage Tank
UFSAR Section 14.5.2	Design Basis Accident
UFSAR Section 14.5.5	Containment Transient Analysis

3.5 RESIDUAL HEAT REMOVAL SYSTEM

Applicability

Applies to the operational status of the Residual Heat Removal System.

Objective

To define the limiting conditions for operation that are necessary to remove decay heat from the Reactor Coolant System in normal shutdown situations.

Specification

A. The reactor shall not be made critical unless:

1. Two residual heat removal pumps are operable.
2. Two residual heat exchangers are operable.
3. All system piping and valves, required to establish a flow path to and from the above components, are operable.
4. All Component Cooling System piping and valves, required to establish a flow path to and from the above components, are operable.

B. The requirements of Specification A may be modified to allow one of the following components (including associated valves and piping) to be inoperable at any one time. If the system is not restored to meet the requirements of Specification A within 14 days, the reactor shall be shutdown.

1. One residual heat removal pump may be out of service, provided immediate attention is directed to making repairs.
2. One residual heat removal heat exchanger may be out of service, provided immediate attention is directed to making repairs.

Basis

The Residual Heat Removal System is required to bring the Reactor Coolant System from conditions of approximately 350°F and pressures between 400 and 450 psig to cold shutdown conditions. Heat removal at greater temperatures is by the Steam and Power Conversion System. The Residual Heat Removal System is provided with two pumps and two heat exchangers. If one of the two pumps and/or one of the two heat exchangers is not operative, safe operation of the unit is not affected; however, the time for cooldown to cold shutdown conditions is extended.

The NRC requires that the series motorized valves in the line connecting the RHRS and RCS be provided with pressure interlocks to prevent them from opening when the reactor coolant system is at pressure.

References

FSAR Section 9.3 - Residual Heat Removal System.

3.6 TURBINE CYCLE

Applicability

Applies to the operating status of the Main Steam and Auxiliary Feed Systems.

Objectives

To define the conditions required in the Main Steam System and Auxiliary Feed System for protection of the steam generator and to assure the capability to remove residual heat from the core during a loss of station power/or accident situations.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not exceed 350°F or 450 psig, respectively, or the reactor shall not be critical unless the five main steam line code safety valves associated with each steam generator in unisolated reactor coolant loops are OPERABLE with lift settings as specified in Table 3.6-1A and 3.6-1B.

B. To assure residual heat removal capabilities, the following conditions shall be met prior to the commencement of any unit operation that would establish reactor coolant system conditions of 350°F and 450 psig which would preclude operation of the Residual Heat Removal System. The following shall apply:

1. Two motor driven auxiliary feedwater pumps shall be OPERABLE.

2. A minimum of 96,000 gallons of water shall be available in the protected condensate storage tank to supply emergency water to the auxiliary feedwater pump suctions.

3. All main steam line code safety valves, associated with steam generators in unisolated reactor coolant loops, shall be OPERABLE with lift settings as specified in Table 3.6-1A and 3.6-1B.

4. The auxiliary feedwater cross-connect capability shall be available, as follows:

   a. Two of the three auxiliary feedwater pumps on the opposite unit (automatic initiation instrumentation need not be OPERABLE) capable of being used with the opening of the cross-connect.

   b. A minimum of 60,000 gallons of water available in the protected condensate storage tank of the opposite unit to supply emergency water to the auxiliary feedwater pump suction of that unit.

   c. Emergency power supplied to the opposite unit's auxiliary feedwater pumps and to the AFW cross-connect valves, as follows:

      1. Two diesel generators (the opposite unit's diesel generator and the shared backup diesel generator) OPERABLE with each generator's day tank having at least 290 gallons of fuel and with a minimum on-site supply of 35,000 gallons of fuel available.

      2. Two 4160V emergency buses energized.

      3. Two OPERABLE flow paths for providing fuel to the opposite unit's diesel generator and the shared backup diesel generator.

      4. Two station batteries, two chargers and the DC distribution systems OPERABLE.

      5. Emergency diesel generator battery, charger and the DC control circuitry OPERABLE for the opposite unit's diesel generator and for the shared back-up diesel generator.

      6. The 480V emergency buses energized which supply power to the auxiliary feedwater cross-connect valves:

         a. For AFW from Unit 1 to Unit 2: Buses 1H1 and 1J1.
         b. For AFW from Unit 2 to Unit 1: Buses 2H1 and 2J1.

7. One of the two physically independent circuits from the offsite transmission network energizing the opposite unit's emergency buses.

C. Prior to reactor power exceeding 10%, the steam driven auxiliary feedwater pump shall be OPERABLE.

D. System piping, valves, and control board indication required for operation of the components enumerated in Specifications 3.6.B and 3.6.C shall be OPERABLE (automatic initiation instrumentation associated with the opposite unit's auxiliary feedwater pumps need not be OPERABLE).

E. The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/cc DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

F. With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two motor driven feedwater pumps and one steam driven feedwater pump) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the following 12 hours.

G. The requirements of Specifications 3.6.B and 3.6.D above concerning the opposite unit's auxiliary feedwater pumps; associated piping, valves, and control board indication; and the protected condensate storage tank may be modified to allow the following components to be inoperable, provided immediate attention is directed to making repairs.

1. One train of the opposite unit's piping, valves, and control board indications or two of the opposite unit's auxiliary feedwater pumps may be inoperable for a period not to exceed 14 days.

2. Both trains of the opposite unit's piping, valves, and control board indications; the opposite unit's protected condensate storage tank; the cross-connect piping from the opposite unit; or three of the opposite unit's auxiliary feedwater pumps may be inoperable for a period not to exceed 72 hours.

3. A train of the opposite unit's emergency power system as required by Section 3.6.B.4.c above may be inoperable for a period not to exceed 14 days: if this train's inoperability is related to a diesel fuel oil path, one diesel fuel oil path may be "inoperable" for 24 hours provided the other flow path is proven OPERABLE: if after 24 hours, the inoperable flow path cannot be restored to service, the diesel shall be considered "inoperable". During this 14 day period, the following limitations apply:

   a. If the offsite power source becomes unable to energize the opposite unit's OPERABLE train, operation may continue provided its associated emergency diesel generator is energizing the OPERABLE train.

   b. If the opposite unit's OPERABLE train's emergency diesel generator becomes unavailable, operation may continue for 72 hours provided the offsite power source is energizing the opposite unit's OPERABLE train.

   c. Return of the originally inoperable train to OPERABLE status allows the second inoperable train to revert to the 14 day limitation.

If the above requirements are not met, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.

H. The requirements of Specification 3.6.B.2 above may be modified to allow utilization of protected condensate storage tank water with the auxiliary steam generator feed pumps provided the water level is maintained above 60,000 gallons, sufficient replenishment water is available in the 300,000 gallon condensate storage tank, and replenishment of the protected condensate storage tank is commenced within two hours after the cessation of protected condensate storage tank water consumption.

Basis

A reactor which has been shutdown from power requires removal of core residual heat. While reactor coolant temperature or pressure is > 350°F or 450 psig, respectively, residual heat removal requirements are normally satisfied by steam bypass to the condenser. If the condenser is unavailable, steam can be released to the atmosphere through the safety valves or power operated relief valves.

The capability to supply feedwater to the generators is normally provided by the operation of the Condensate and Feedwater Systems. In the event of complete loss of electrical power to the station. residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps and the 110,000-gallon protected condensate storage tank.

In the event of a fire or high energy line break which would render the auxiliary feedwater pumps inoperable on the affected unit, residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor-driven auxiliary feedwater pumps from the opposite unit. A minimum of two auxiliary feedwater pumps are required to be operable* on the opposite unit to ensure compliance with the design basis accident analysis assumptions, in that auxiliary feedwater can be delivered via the cross-connect, even if a single active failure results in the loss of one of the two pumps. In addition, the requirement for operability of the opposite unit's emergency power system is to ensure that auxiliary feedwater from the opposite unit can be supplied via the cross-connect in the event of a common-mode failure of all auxiliary feedwater pumps in the affected unit due to a high energy line break in the main steam valve house. Without this requirement, a single failure (such as loss of the shared backup diesel generator) could result in loss of power to the opposite unit's emergency buses in the event of a loss of offsite power, thereby rendering the cross-connect inoperable. The longer allowed outage time for the opposite unit's emergency power system is based on the low probability of a high energy line break in the main steam valve house coincident with a loss of offsite power.

* excluding automatic initiation instrumentation

The specified minimum water volume in the 110,000-gallon protected condensate storage tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all offsite electrical power. It is also sufficient to maintain one unit at hot shutdown for 2 hours, followed by a 4 hour cooldown from 547°F to 350°F (i.e., RHR operating conditions). If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000-gallon condensate tank can be gravity-fed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suctions is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,842,454 pounds per hour at their individual relieving pressure; the total combined capacity of all fifteen main steam code safety valves is 11,527,362 pounds per hour. The nominal power rating steam flow is 11,260,000 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady state power than can be obtained during three reactor coolant loop operation.

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine - 131 activity is based on limiting the inhalation dose at the site boundary following a postulated steam line break accident to a small fraction of the 10 CFR 100 limits. The accident analysis, which is performed based on the guidance of NUREG-0800 Section 15.1-5, assumes the release of the entire contents of the faulted steam generator to the atmosphere.

REFERENCES

FSAR Section 4, Reactor Coolant System

FSAR Section 9.3, Residual Heat Removal System

FSAR Section 10.3.1, Main Steam System

FSAR Section 10.3.2, Auxiliary Steam System

FSAR Section 10.3.5, Auxiliary Feedwater System

FSAR Section 10.3.8, Vent and Drain Systems

FSAR Section 14.3.2.5, Environmental Effects of a Steam Line Break

TABLE 3.6-1A

UNIT 1
MAIN STEAM SAFETY VALVE LIFT SETTING

VALVE NUMBER	LIFT SETTING *#	ORIFICE SIZE
SV-MS-101A, B, C	1085 psig	7.07 sq. in.
SV-MS-102A, B, C	1095 psig	16 sq. in.
SV-MS-103A, B, C	1110 psig	16 sq. in.
SV-MS-104A, B, C	1120 psig	16 sq. in.
SV-MS-105A, B, C	1135 psig	16 sq. in.

TABLE 3.6-1B

UNIT 2
MAIN STEAM SAFETY VALVE LIFT SETTING

VALVE NUMBER	LIFT SETTING *#	ORIFICE SIZE
SV-MS-201A, B, C	1085 psig	7.07 sq. in.
SV-MS-202A, B, C	1095 psig	16 sq. in.
SV-MS-203A, B, C	1110 psig	16 sq. in.
SV-MS-204A, B, C	1120 psig	16 sq. in.
SV-MS-205A, B, C	1135 psig	16 sq. in.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

\# The as found condition shall be ± 3% and the as left condition shall be ± 1%.

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to reactor and safety features instrumentation systems.

Objectives

To ensure the automatic initiation of the Reactor Protection System and the Engineered Safety Features in the event that a principal process variable limit is exceeded. and to define the limiting conditions for operation of the plant instrumentation and safety circuits necessary to ensure reactor and plant safety.

Specification

A. The Reactor Protection System instrumentation channels and interlocks shall be OPERABLE as specified in Table 3.7-1.

B. The Engineered Safeguards Actions and Isolation Function Instrumentation channels and interlocks shall be OPERABLE as specified in Tables 3.7-2 and 3.7-3, respectively.

C. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.7-4.

D. The explosive gas monitoring instrumentation channel shown in Table 3.7-5(a) shall be OPERABLE with its alarm setpoint set to ensure that the limits of Specification 3.11.A.1 are not exceeded.

1. With an explosive gas monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, declare the channel inoperable and take the action shown in Table 3.7-5(a).

2. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the action shown in Table 3.7-5(a). Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission (Region II) to explain why the inoperability was not corrected in a timely manner.

E. The accident monitoring instrumentation listed in Table 3.7-6 shall be OPERABLE in accordance with the following:

1. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum OPERABLE Channels requirement of Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

F. The containment hydrogen analyzers and associated support equipment shall be OPERABLE in accordance with the following:

1. Two independent containment hydrogen analyzers shall be OPERABLE during REACTOR CRITICAL or POWER OPERATION.

   a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one analyzer to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 6 hours.

NOTE: Operability of the hydrogen analyzers includes proper operation of the respective Heat Tracing System.

Basis

Instrument Operating Conditions

During plant operations, the complete instrumentation system 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 the limiting conditions for operation necessary to preserve the effectiveness of the Reactor Protection System when any one or more of the channels is out of service.

Almost all Reactor Protection System 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 Nuclear Instrumentation System (NIS) channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the dropped-rod protection from NIS, for the channel being tested, (b) placing the $\Delta T/T_{avg}$ protection channel set that is being fed from the NIS channel in the trip mode, and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features.(1)

Safety Injection System Actuation

Protection against a loss-of-coolant or steam line break accident is provided by automatic actuation of the Safety Injection System (SIS) which provides emergency cooling and reduction of reactivity.

The loss-of-coolant accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The engineered safeguards instrumentation has been designed to sense these effects of the loss-of-coolant accident by detecting low pressurizer pressure to generator signals actuating the SIS active phase. The SIS active phase is also actuated by a high containment pressure signal brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between the steam header and steam generator line or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason, protection against a steam line break accident is also provided by low pressurizer pressure actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features also initiate containment spray upon sensing a high-high containment pressure signal. The containment spray acts to reduce containment pressure in the event of a loss-of-coolant or steam line break accident inside the containment. The containment spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure than the SIS. Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high-high containment pressure sensed by 3 out of the 4 containment pressure signals.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing the steam line trip valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high-high containment pressure or high steam line flow with coincident low steam line pressure or low reactor coolant average temperature. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the SIS in order to prevent excessive cooldown of the Reactor Coolant System. This mitigates the effects of an accident such as a steam line break which in itself causes excessive coolant temperature cooldown. Feedwater line isolation also reduces the consequences of a steam line break inside the containment by stopping the entry of feedwater.

Auxiliary Feedwater System Actuation

The automatic initiation of auxiliary feedwater flow to the steam generators by instruments identified in Table 3.7-2 ensures that the Reactor Coolant System decay heat can be removed following loss of main feedwater flow. This is consistent with the requirements of the "TMI-2 Lessons Learned Task Force Status Report," NUREG-0578, item 2.1.7.b.

Setting Limits

1. The high containment pressure limit is set at about 10% of design containment pressure. Initiation of safety injection protects against loss of coolant(2) or steam line break(3) accidents as discussed in the safety analysis.
2. The high-high containment pressure limit is set at about 23% of design containment pressure. Initiation of containment spray and steam line isolation protects against large loss-of-coolant(2) or steam line break accidents(3) as discussed in the safety analysis.
3. The pressurizer low pressure setpoint for safety injection actuation 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.(2) The setting limit (in units of psig) is based on nominal atmospheric pressure.
4. The steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis.(3)
5. The high steam line flow differential pressure setpoint is constant at 40% full flow between no load and 20% load and increasing linearly to 110% of full flow at full load in order to protect against large steam line break accidents. The coincident low T_{avg} setting limit for SIS and steam line isolation initiation is set below its HOT SHUTDOWN value. The coincident

steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.(3)

Accident Monitoring Instrumentation

The operability of the accident monitoring instrumentation in Table 3.7-6 ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. On the pressurizer PORVs, the pertinent channels consist of redundant limit switch indication. The pressurizer safety valves utilize an acoustic monitor channel and a downstream high temperature indication channel. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations." Potential gaseous effluent release paths are equipped with radiation monitors to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. The gaseous effluent release paths monitored are the process vent stack, ventilation vent stack, main steam safety valve and atmospheric dump valve discharge and the AFW pump turbine exhaust. The potential liquid effluent release paths via the service water discharge from the recirculation spray heat exchangers are equipped with radiation monitors to detect leakage of recirculated containment sump fluid. These radiation monitors and the associated sample pumps are required to operate during the recirculation heat removal phase following a loss of coolant accident in order to detect a passive failure of a recirculation spray heat exchanger tube. These monitors meet the requirements of NUREG-0737.

Instrumentation is provided for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the Waste Gas Holdup System. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Containment Hydrogen Analyzers

Indication of hydrogen concentration in the containment atmosphere can be provided in the control room over the range of zero to ten percent hydrogen concentration under accident conditions.

These redundant, qualified analyzers are shared by Units 1 and 2 with instrumentation to indicate and record the hydrogen concentration. Each hydrogen analyzer is designed with the capability to obtain an accurate sample within 30 minutes after initiation of safety injection.

A transfer switch is provided for Unit 1 to use both analyzers or for Unit 2 to use both analyzers. In addition, each unit's hydrogen analyzer has a transferable emergency power supply from Unit 1 and Unit 2. This will ensure redundancy for each unit.

Indication of Unit 1 and Unit 2 hydrogen concentration is provided on the Unit 1 Post Accident Monitoring panel and the Unit 2 Post Accident Monitoring panel, respectively. Hydrogen concentration is also recorded on qualified recorders. In addition, each hydrogen analyzer is provided with an alarm for trouble/high hydrogen content. These alarms are located in the control room.

The supply lines installed from the containment penetrations to the hydrogen analyzers have Category I Class IE heat tracing applied. The heat tracing system receives the same transferable emergency power as is provided to the containment hydrogen analyzers. The heat trace system is de-energized during normal system operation. Upon receipt of a SIS, after a preset time delay, heat tracing is energized to bring the piping process temperature to 250 ± 10°F. Each heat trace circuit is equipped with an RTD to provide individual circuit readout, over-temperature alarm, and control the circuit to maintain the process temperatures.

The hydrogen analyzer heat trace system is equipped with high temperature, loss of D.C. power, loss of A.C. power, loss of control power, and failure of automatic initiation alarms.

<u>Non-Essential Service Water Isolation System</u>

The operability of this functional system ensures that adequate intake canal inventory can be maintained by the Emergency Service Water Pumps. Adequate intake canal inventory provides design service water flow to the recirculation spray heat exchangers and other essential loads (e.g., control room area chillers, charging pump lube oil coolers) following a design basis loss of coolant accident with a coincident loss of offsite power. This system is common to both units in that each of the two trains will actuate equipment on each unit.

Clarification of Operator Actions

The Operator Actions associated with Functional Units 10 and 16 on Table 3.7-1 require the unit to be reduced in power to less than the P-7 setpoint (10%) if the required conditions cannot be satisfied for either the P-8 or P-7 permissible bypass conditions. The requirement to reduce power below P-7 for a P-8 permissible bypass condition is necessary to ensure consistency with the out of service and shutdown action times assumed in the WCAP-10271 and WCAP-14333P risk analyses by eliminating the potential for a scenario that would allow sequential entry into the Operator Actions (i.e.. initial entry into the Operator Action with a reduction in power to below P-8. followed by a second entry into the Operator Action with a reduction in power to below P-7). This scenario would permit sequential allowed outage time periods that may result in an additional 72 hours that was not assumed in the risk analysis to place a channel in trip or to place the unit in a condition where the protective function was not necessary.

References

(1) UFSAR - Section 7.5

(2) UFSAR - Section 14.5

(3) UFSAR - Section 14.3.2

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
1. Manual	2	2	1		1
2. Nuclear Flux Power Range	4	3	2	Low trip setting at P-10	2
3. Nuclear Flux Intermediate Range	2	2	1	P-10	3
4. Nuclear Flux Source Range				P-6	
a. Below P-6 - Note A	2	2	1		4
b. Shutdown - Note B	2	1	0		5
5. Overtemperature ΔT	3	2	2		6
6. Overpower ΔT	3	2	2		6
7. Low Pressurizer Pressure	3	2	2	P-7	7
8. Hi Pressurizer Pressure	3	2	2		6

Note A - With the reactor trip breakers closed and the control rod drive system capable of rod withdrawal.

Note B - With the reactor trip breakers open.

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
9. Pressurizer-Hi Water Level	3	2	2	P-7	7
10. Low Flow	3/loop	2/loop in each operating loop	2/loop in any operating loop	P-8	7
			2/loop in any 2 operating loops	P-7	7
11. Turbine Trip					
a. Stop valve closure	4	1	4	P-7	7
b. Low fluid oil pressure	3	2	2	P-7	7
12. Lo-Lo Steam Generator Water Level	3/loop	2/loop in each operating loop	2/loop in any operating loops		6
13. Underfrequency 4KV Bus	3-1/bus	2	2	P-7	7
14. Undervoltage 4KV Bus	3-1/bus	2	2	P-7	7
15. Safety Injection (SI) Input From ESF	2	2	1		11
16. Reactor Coolant Pump Breaker Position	1/breaker	1/breaker per operating loop	1 2	P-8 P-7	9 9

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
17. Low steam generator water level with steam/feedwater flow mismatch	2/loop-level and 2/loop-flow mismatch	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop		6
18. a. Reactor Trip Breakers	2	2	1		8
b. Reactor Trip Bypass Breakers - Note C	2	1	1		
19. Automatic Trip Logic	2	2	1		11
20. Reactor Trip System Interlocks - Note D					
a. Intermediate range neutron flux, P-6	2	2	1		13
b. Low power reactor trips block, P-7					
Power range neutron flux, P-10	4	3	2		13
and					
Turbine impulse pressure	2	2	1		13
c. Power range neutron flux, P-8	4	3	2		13
d. Power range neutron flux, P-10	4	3	2		13
e. Turbine impulse pressure	2	2	1		13

Note C - With the Reactor Trip Breaker open for surveillance testing in accordance with Specification Table 4.1-1 (Item 30)

Note D - Reactor Trip System Interlocks are described in Table 4.1-A

TABLE 3.7-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

ACTION 1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN and open the reactor trip breakers within the next 6 hours.

ACTION 2. With the number of OPERABLE channels equal to the Minimum OPERABLE Channels requirement, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.
2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of the redundant channel(s) per Specification 4.1.
3. Either, THERMAL POWER is restricted to ≤ 75% of RATED POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED POWER within 78 hours; or, the QUADRANT POWER TILT is monitored at least once per 12 hours.

TABLE 3.7-1 (Continued)

4. The QUADRANT POWER TILT shall be determined to be within the limit when above 75 percent of RATED POWER with one Power Range Channel inoperable by using the moveable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT at least once per 12 hours.

With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours

ACTION 3. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below the P-6 (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, but below 10% of RATED POWER, decrease power below P-6 or, increase THERMAL POWER above 10% of RATED POWER within 24 hours.

c. Above 10% of RATED POWER, POWER OPERATION may continue.

TABLE 3.7-1 (Continued)

ACTION 4. With the number of channels OPERABLE one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below P-6, (Block of Source Range Reactor Trip) setpoint, immediately suspend reactivity changes that are more positive than necessary to meet the required shutdown margin or refueling boron concentration limit and restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. With two Source Range Channels inoperable, open the reactor trip breakers immediately. Two Source Range channels must be OPERABLE prior to increasing THERMAL POWER above the P-6 setpoint.

b. Above P-6, operation may continue.

ACTION 5. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, verify compliance with the Shutdown Margin requirements within 1 hour and at least once per 12 hours thereafter.

ACTION 6. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied in the time permitted, be in at least HOT SHUTDOWN within 6 hours.

TABLE 3.7-1 (Continued)

ACTION 7. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours.

ACTION 8.A. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE, or one reactor trip breaker may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor trip breaker and automatic trip logic provided the other train is OPERABLE.

8.B. With one of the diverse trip features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply Action 8.A. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.7-1 (Continued)

ACTION 9. With one channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or reduce THERMAL POWER to below the P-7 (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) setpoint within the next 6 hours.

ACTION 10. Deleted

ACTION 11. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 12. Deleted

ACTION 13. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-2

ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
1. SAFETY INJECTION (SI)					
a. Manual	2	2	1		21
b. High containment pressure	4	3	3		17
c. High differential pressure between any steam line and the steam header	3/steam line	2/steam line	2/steam line on any steam line	Primary pressure less than 2000 psig, except when reactor is critical	20
d. Pressurizer low-low pressure	3	2	2	Primary pressure less than 2000 psig, except when reactor is critical	20
e. High steam flow in 2/3 steam lines coincident with low T_{avg} or low steam line pressure					
1) Steam line flow	2/steam line	1/steam line	1/steam line any two lines	Reactor coolant T_{avg} less than 543° during heatup and cooldown	20
2) T_{avg}	1/loop	1/loop any two loops	1/loop any two loops	Reactor coolant T_{avg} less than 543° during heatup and cooldown	20
3) Steam line pressure	1/line	1/line any two loops	1/line any two loops	Reactor coolant T_{avg} less than 543° during heatup and cooldown	20
f. Automatic actuation logic	2	2	1		14

TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
2. CONTAINMENT SPRAY					
a. Manual	1 set	1 set	1 set♦		15
b. High containment pressure (Hi-Hi)	4	3	3		17
c. Automatic actuation logic	2	2	1		14
3. AUXILIARY FEEDWATER					
a. Steam generator water level low-low					
1) Start motor driven pumps	3/steam generator	2/steam generator	2/steam generator any 1 generator		20
2) Starts turbine driven pump	3/steam generator	2/steam generator	2/steam generator any 2 generators		20
b. RCP undervoltage starts turbine driven pump	3	2	2		20
c. Safety injection - start motor driven pumps	See #1 above (all SI initiating functions and requirements)				
d. Station blackout - start motor driven pumps	1/bus 2 transfer buses/unit	1/bus 2 transfer buses/unit	2		24

♦ Must actuate 2 switches simultaneously

TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
3. AUXILIARY FEEDWATER (continued)					
e. Trip of main feedwater pumps - start motor driven pumps	2/MFW pump	1/MFW pump	2-1 each MFW pump		24
f. Automatic actuation logic	2	2	1		22
4. LOSS OF POWER					
a. 4.16 kv emergency bus undervoltage (loss of voltage)	3/bus	2/bus	2/bus		26
b. 4.16 kv emergency bus undervoltage (degraded voltage)	3/bus	2/bus	2/bus		26
5. NON-ESSENTIAL SERVICE WATER ISOLATION					
a. Low intake canal level	4	3	3		20
b. Automatic actuation logic	2	2	1		14
6. ENGINEERED SAFEGUARDS ACTUATION INTERLOCKS - Note A					
a. Pressurizer pressure, P-11	3	2	2		23
b. Low-low T_{avg}, P-12	3	2	2		23
c. Reactor trip, P-4	2	2	1		24
7. RECIRCULATION MODE TRANSFER					
a. RWST Level - Low	4	3	2		25
b. Automatic Actuation Logic and Actuation Relays	2	2	1		14

Note A - Engineered Safeguards Actuation Interlocks are described in Table 4.1-A

Revised by NRC letter dated September 3, 2002

TABLE 3.7-3
INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
1. CONTAINMENT ISOLATION					
a. Phase 1					
1) Safety Injection (SI)	See Item #1, Table 3.7-2 (all SI initiating functions and requirements)				
2) Automatic initiation logic	2	2	1		14
3) Manual	2	2	1		21
b. Phase 2					
1) High containment pressure	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	2	2	1		21
c. Phase 3					
1) High containment pressure (Hi-Hi setpoint)	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	1 set	1 set	1 set♦		15
2. STEAMLINE ISOLATION					
a. High steam flow in 2/3 lines coincident with 2/3 low T_{avg} or 2/3 low steam pressures	See Item #1.e Table 3.7-2 for operability requirements				

♦ Must actuate 2 switches simultaneously

TABLE 3.7-3 (Continued)
INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
STEAMLINE ISOLATION (continued)					
b. High containment pressure (Hi-Hi setpoint)	4	3	3		17
c. Manual	1/steamline	1/steamline	1/steamline		21
d. Automatic actuation logic	2	2	1		22
3. TURBINE TRIP AND FEEDWATER ISOLATION				When all MFRV, SG FWIV & associated bypass valves are closed & deactivated or isolated by manual valves.	
a. Steam generator water-level high-high	3/steam generator	2/steam generator	2/in any one steam generator		20
b. Automatic actuation logic and actuation relay	2	2	1		22
c. Safety injection	See Item #1 Table 3.7-2 (all SI initiating functions and requirements)				

TABLES 3.7-2 AND 3.7-3

TABLE NOTATIONS

ACTION 14. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the next 30 hours. One channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 15. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 16. Deleted

ACTION 17. With the number of OPERABLE channels one less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 72 hours and the Minimum OPERABLE Channels requirement is met. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

ACTION 18. Deleted

ACTION 19. Deleted

ACTION 20. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and reduce RCS temperature & pressure to less than 350°F/450 psig, respectively in the following 12 hours.

TABLES 3.7-2 ANDS 3.7-3 (Continued)

TABLE NOTATIONS

ACTION 21. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 22. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 6 hours and reduce pressure and temperature to less than 450 psig and 350° within the following 12 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 23. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 24. With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or reduce pressure and temperature to less than 450 psig and 350°F within the next 12 hours.

ACTION 25. With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the bypassed condition within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

ACTION 26. With the number of OPERABLE channels less than the Total Number of Channels, the associated Emergency Diesel Generator may be considered OPERABLE provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped conditions within 72 hours.

b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied, declare the associated EDG inoperable.

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	Functional Unit	Channel Action	Setting Limit
1	High Containment Pressure (High Containment Pressure Signal)	a) Safety Injection b) Containment Vacuum Pump Trip c) High Press. Containment Isolation d) Safety Injection Containment Isolation e) F.W. Line Isolation	$\leq$ 19 psia
2	High-High Containment Pressure (High-High Containment Pressure Signals)	a) Containment Spray b) Recirculation Spray c) Steam Line Isolation d) High-High Press. Containment Isolation	$\leq$ 25 psia
3	Pressurizer Low-Low Pressure	a) Safety Injection b) Safety Injection Containment Isolation c) F.W. Line Isolation	$\geq$ 1,760 psig
4	High Differential Pressure Between Steam Line and the Steam Line Header	a) Safety Injection b) Safety Injection Containment Isolation c) F.W. Line Isolation	$\leq$ 150 psig
5	High Steam Flow in 2/3 Steam Lines	a) Safety Injection	$\leq$ 40% (at zero load) of full steam flow $\leq$ 40% (at 20% load) of full steam flow $\leq$ 110% (at full load) of full steam flow
		b) Steam Line Isolation c) Safety Injection Containment Isolation d) F.W. Line Isolation	
	Coincident with Low T_{avg} or		$\geq$ 541°F T_{avg}
	Low Steam Line Pressure		$\geq$ 500 psig steam line pressure

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	Functional Unit	Channel Action	Setting Limit
6	AUXILIARY FEEDWATER		
	a. Steam Generator Water Level Low-Low	Aux. Feedwater Initiation S/G Blowdown Isolation	≥ 14.5% narrow range
	b. RCP Undervoltage	Aux. Feedwater Initiation	≥ 70% nominal
	c. Safety Injection	Aux. Feedwater Initiation	All S.I. setpoints
	d. Station Blackout	Aux. Feedwater Initiation	≥ 46.7% nominal
	e. Main Feedwater Pump Trip	Aux. Feedwater Initiation	N.A.
7	LOSS OF POWER		
	a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	Emergency Bus Separation and Diesel start	≥ 2975 volts and ≤ 3265 volts with a 2 (+5, -0.1) second time delay
	b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	Emergency Bus Separation and Diesel start	≥ 3830 volts and ≤ 3881 volts with a 60 (±3.0) second time delay (Non CLS, Non SI) 7 (±0.35) second time delay (CLS or SI Conditions)
8	NON-ESSENTIAL SERVICE WATER ISOLATION		
	a. Low Intake Canal Level	Isolation of Service Water flow to non-essential loads	23 feet-6 inches
9	RECIRCULATION MODE TRANSFER		
	a. RWST Level-Low	Initiation of Recirculation Mode Transfer System	≥ 11.25% ≤ 15.75%
10.	TURBINE TRIP AND FEEDWATER ISOLATION		
	a. Steam Generator Water Level High-High	Turbine Trip Feedwater Isolation	≤ 80% narrow range

TABLE 3.7-5
AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

Monitor Channel	Automatic Function At Alarm Conditions	Monitoring Requirements	Alarm Setpoint µ CI/cc
1. Component cooling water radiation monitors	Shuts surge tank vent valve HCV-CC-100	See Specification 3.13	Twice Background

TABLE 3.7-5(a)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

Instrument	Total No. of Channels	Minimum OPERABLE Channels	Action
1. Waste Gas Holdup System Explosive Gas Monitoring System Oxygen Monitor	1	1	1

ACTION 1 - With the number of channels OPERABLE less than required by the minimum OPERABLE channels requirement, operation of this waste gas holdup system may continue provided grab samples are collected (1) at least once per 4 hours during degassing operations to the waste gas decay tank and (2) at least once per 24 hours during other operations. Samples shall be analyzed within 4 hours after collection.

TABLE 3.7-6

ACCIDENT MONITORING INSTRUMENTATION

	Instrument	Total No. Of Channels	Minimum OPERABLE Channels
1.	Auxiliary Feedwater Flow Rate	1 per S/G	1 per S/G
2.	Inadequate Core Cooling Monitor		
	a. Reactor Vessel Coolant Level Monitor	2	1
	b. Reactor Coolant System Subcooling Margin Monitor	2	1
	c. Core Exit Thermocouples	2 (Note 2)	1 (Note 2)
3.	PORV Position Indicator	2/valve	1/valve
4.	PORV Block Valve Position Indicator	1/valve	1/valve
5.	Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve
6.	Safety Valve Position Indicator (Backup Detector)	1/valve	0
7.	Containment Pressure	2	1
8.	Containment Water Level (Narrow Range)	2	1
9.	Containment Water Level (Wide Range)	2	1
10.	Containment High Range Radiation Monitor	2	1 (Note 1, b and c only)
11.	Process Vent High Range Effluent Monitor	2	2 (Note 1, a, b, and c)
12.	Ventilation Vent High Range Effluent Monitor	2	2 (Note 1, a, b, and c)
13.	Main Steam High Range Radiation Monitors (Units 1 and 2)	3	3 (Note 1, a, b, and c)
14.	Aux. Feed Pump Steam Turbine Exhaust Radiation Monitor	1	1 (Note 1, a, b, and c)
15.	Recirculation Spray Heat Exchanger Service Water Outlet Radiation Monitors	1 per RSHX	1 per RSHX (Note 1, a, b, and c)

Note 1: With the number of operable channels less than required by the Minimum OPERABLE Channels requirements

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours

b. Either restore the inoperable channel to operable status within 7 days of the event, or

c. Prepare and submit a Special Report to the commission pursuant to specification 6.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable.

Note 2: A minimum of 2 core exit thermocouples per quadrant are required for the channel to be operable.

3.8 CONTAINMENT

Applicability

Applies to the integrity and operating pressure of the reactor containment.

Objective

To define the limiting operating conditions of the reactor containment.

Specification

A. CONTAINMENT INTEGRITY

1. CONTAINMENT INTEGRITY, as defined in TS Section 1.0, shall be maintained whenever the Reactor Coolant System temperature exceeds 200°F.

   a. Without CONTAINMENT INTEGRITY, re-establish CONTAINMENT INTEGRITY in accordance with the definition within 1 hour.

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

2. The inside and outside isolation valves in the Containment Ventilation Purge System shall be locked, sealed, or otherwise secured closed whenever the Reactor Coolant System temperature exceeds 200°F.

3. The inside and outside isolation valves in the containment vacuum ejector suction line shall be locked, sealed, or otherwise secured closed whenever the Reactor Coolant System temperature exceeds 200°F.

B. Containment Airlocks

1. Each containment airlock shall be OPERABLE with both doors of the personnel airlock closed except when the airlock is being used for normal transit entry and exit through the containment, then at least one airlock door shall be closed.

   a. With one airlock or associated interlock inoperable, maintain the OPERABLE door closed and either restore the inoperable door to OPERABLE status or lock closed the OPERABLE door within 24 hours.

   b. If the personnel airlock inner door or interlock is inoperable, the outer personnel airlock door may be opened for repair and retest of the inner door. If the inoperability is due to the personnel airlock inner door seal exceeding the leakage test acceptance criteria, the outer personnel airlock door may be opened for a period of time not to exceed fifteen minutes with an annual cumulative time not to exceed one hour per year for repair and retest of the inner door seal.

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

C. Containment Isolation Valves

1. Containment isolation valves shall be OPERABLE.† With one or more isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE† in each affected penetration that is open and either:

   a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

† Non-automatic or deactivated automatic containment isolation valves may be opened on an intermittent basis under administrative control. The valves identified in TS 3.8.A.2 and TS 3.8.A.3 are excluded from this provision.

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Otherwise, place the unit in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

D. Internal Pressure

1. Containment air partial pressure shall be maintained within the acceptable operation range as identified in Figure 3.8-1 whenever the Reactor Coolant System temperature and pressure exceed 350°F and 450 psig, respectively.

   a. With the containment air partial pressure outside the acceptable operation range, restore the air partial pressure to within acceptable limits within 1 hour or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Basis

CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment will be restricted to those leakage paths and associated leak rates assumed in the accident analysis. These restrictions, in conjunction with the allowed leakage, will limit the site boundary radiation dose to within the limits of 10 CFR 100 during accident conditions.

The operability of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. The opening of manual or deactivated automatic containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and

(3) assuring that environmental conditions will not preclude access to close the valves and 4) that this administrative or manual action will prevent the release of radioactivity outside the containment.

The Reactor Coolant System temperature and pressure being below 350°F and 450 psig, respectively, ensures that no significant amount of flashing steam will be formed and hence that there would be no significant pressure buildup in the containment if there is a loss-of-coolant accident. Therefore, the containment internal pressure is not required to be subatmospheric prior to exceeding 350°F and 450 psig.

The allowable value for the containment air partial pressure is presented in TS Figure 3.8-1 for service water temperatures from 25 to 95°F. The RWST water shall have a maximum temperature of 45°F.

The horizontal limit line in TS Figure 3.8-1 is based on LOCA peak calculated pressure criteria, and the sloped line is based on LOCA subatmospheric peak pressure criteria.

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize to 0.5 psig within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident.

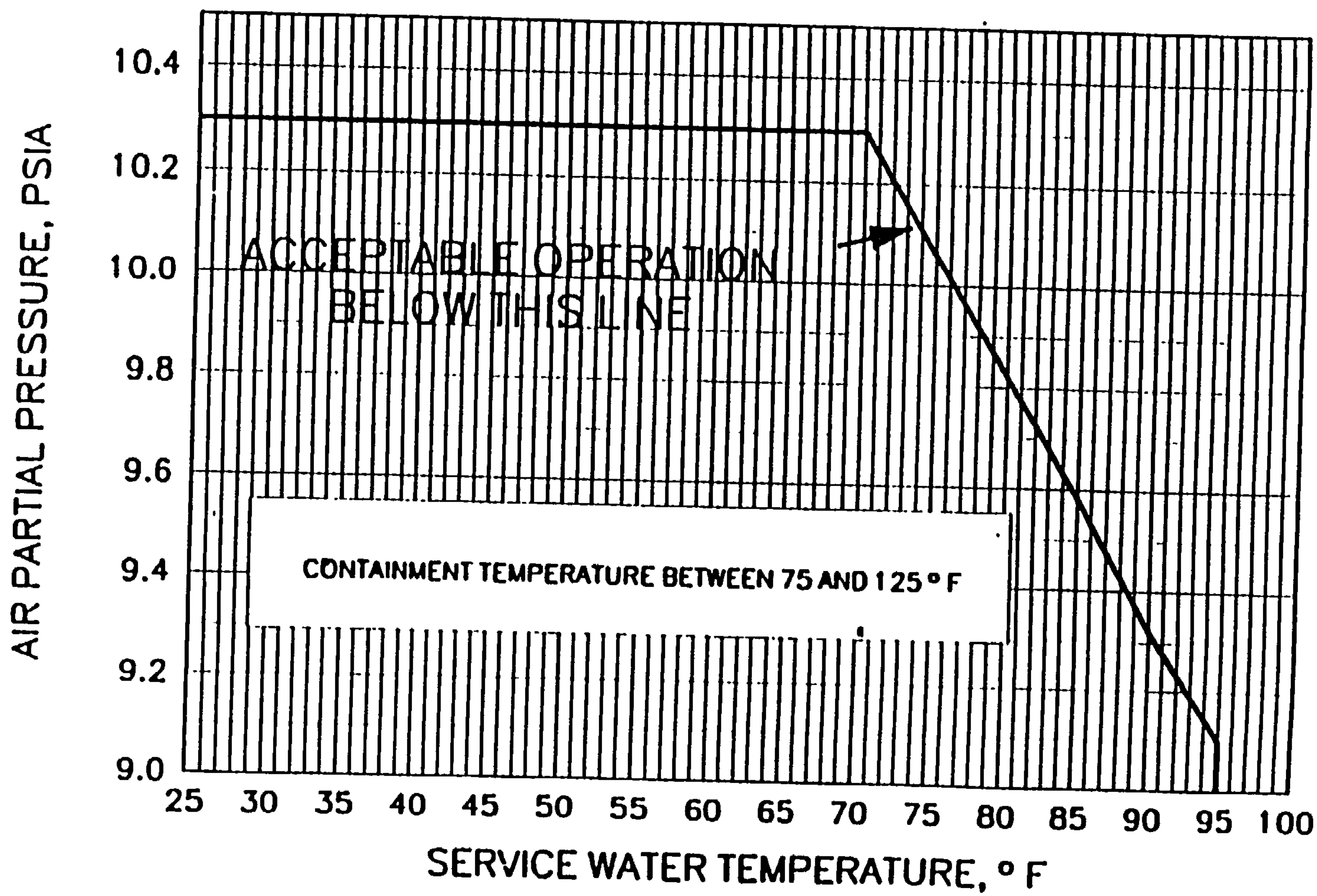
If the containment air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the HOT SHUTDOWN condition. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia, and the bottom mat liner is capable of withstanding an internal pressure as low as 8 psia.

References

UFSAR Section 4.3.2	Reactor Coolant Pump
UFSAR Section 5.2	Containment Isolation
UFSAR Section 5.2.1	Design Bases
UFSAR Section 5.5.2	Isolation Design
UFSAR Section 6.3.2	Containment Vacuum System

TS FIGURE 3.8–1

SURRY TECHNICAL SPECIFICATION CURVE
MAX CONTAINMENT ALLOWABLE AIR PARTIAL PRESSURE INDICATION VS. SW TEMP



Amendment Nos. 203 and 203
AUG 3 1995

3.9 STATION SERVICE SYSTEMS

Applicability

Applies to availability of electrical power for operation of station auxiliaries.

Objective

To define those conditions of electrical power availability necessary to provide for safe reactor operation.

Specification

A. A unit's reactor shall not be made critical without:

1. All three of the unit's 4,160V buses energized

2. All six of the unit's 480V buses energized

3. Both of the 125 V DC buses energized as explained in Section 3.16

4. One battery charger per battery operating as explained in Section 3.16

5. Both of the 4,160V emergency buses energized as explained in Section 3.16

6. All four of the 480V emergency buses energized as explained in Section 3.16

7. Two emergency diesel generators OPERABLE as explained in Section 3.16.

B. The requirements of Specification 3.9-A items 3, 4, 5, 6, and 7 may be modified as provided in Section 3.16-B.

Basis

During startup of a unit, the station's 4,160V and 480V normal and emergency buses are energized from the station's 34.5KV buses. At reactor power levels greater than 5 percent of rated power the 34.5KV buses are required to energize only the emergency buses because at this power level the station generator can supply sufficient power to the normal 4,160V and 480V lines to operate the unit. Three reactor coolant loop operation with all 4,160V and 480V buses energized is the normal mode of operation for a unit.

The electrical power requirements and the emergency power testing requirements for the auxiliary feedwater cross-connect are contained in TS 3.6.B.4.c and TS 4.6, respectively.

References

FSAR Section 8.4 Station Service Systems

FSAR Section 8.5 Emergency Power Systems

3.10 REFUELING

Applicability

Applies to operating limitations during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building.

Objective

To assure that no accident could occur during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building that would affect public health and safety.

Specification

A. During REFUELING OPERATIONS the following conditions are satisfied:

1. The equipment access hatch and at least one door in the personnel airlock shall be capable of being closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the containment isolation valves shall be OPERABLE or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration shall be capable of being closed.

2. At least one source range neutron detector shall be in service at all times when the reactor vessel head is unbolted. Whenever core geometry or coolant chemistry is being changed, subcritical neutron flux shall be continuously monitored by at least two source range neutron detectors, each with continuous visual indication in the Main Control Room and one with audible indication within the containment. During core fuel loading phases, there shall be a minimum neutron count rate detectable on two operating source range neutron detectors with the exception of initial core loading, at which time a minimum neutron count rate need be established only when there are eight (8) or more fuel assemblies loaded into the reactor vessel.

3. The manipulator crane area monitors and the containment particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

4. At least one residual heat removal pump and heat exchanger shall be OPERABLE to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of core alterations or reactor vessel surveillance inspections.

5. Two residual heat removal pumps and heat exchangers shall be OPERABLE to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

6. At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

7. With the reactor vessel head unbolted or removed, any filled portions of the Reactor Coolant System and the refueling canal shall be maintained at a boron concentration which is:

   a. Sufficient to maintain K-effective equal to 0.95 or less, and

   b. Greater than or equal to 2300 ppm and shall be checked by sampling at least once every 72 hours.

8. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

9. No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

10. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

    This restriction does not apply to the movement of the transfer canal door.

11. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

12. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

B. During irradiated fuel movement in the Fuel Building the following conditions are satisfied:

   1. The fuel pit bridge area monitor and the ventilation vent stack 2 particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

   2. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

      This restriction does not apply to the movement of the transfer canal door.

   3. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

4. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

5. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

C. If any one of the specified limiting conditions for refueling is not met, REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

D. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

E. The requirements of 3.0.1 are not applicable.

Basis

Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

Potential escape paths for fission product radioactivity within containment are required to be closed or capable of closure to prevent the release to the environment. However, since there is no potential for significant containment pressurization during refueling, the Appendix J leakage criteria and tests are not applicable.

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch must be capable of being closed.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during periods when CONTAINMENT INTEGRITY is required. Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors to remain open for extended periods when frequent containment entry is necessary. During REFUELING OPERATIONS, containment closure does not have to be maintained, but airlock doors may need to be closed to establish containment closure. Therefore, the door interlock mechanism may remain disabled, but one airlock door must be capable of being closed.

Containment penetrations that terminate in the Auxiliary Building or Safeguards and provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being closed by at least one barrier during REFUELING OPERATIONS. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by at least one barrier during REFUELING OPERATIONS. Isolation may be achieved by an OPERABLE isolation valve, a closed valve, a blind flange, or by an equivalent isolation method. Equivalent isolation methods must be evaluated and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier.

For the personnel airlock, equipment access hatch, and other penetrations, 'capable of being closed' means the openings are able to be closed; they do not have to be sealed or meet the leakage criteria of TS 4.4. Station procedures exist that ensure in the event of a fuel handling accident, that the open personnel airlock and other penetrations can and will be closed. Closure of the equipment hatch will be accomplished in accordance with station procedures and as allowed by dose rates in containment. The radiological analysis of the fuel handling accident does not take credit for closure of the personnel airlock, equipment access hatch or other penetrations.

The fuel building ventilation exhaust and containment ventilation purge exhaust may be diverted through charcoal filters whenever refueling is in progress. However, there is no requirement for filtration since the Fuel Handling Accident analysis takes no credit for these filters. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

During refueling, the reactor refueling water cavity is filled with approximately 220,000 gal of water borated to at least 2,300 ppm boron. The boron concentration of this water, established by Specification 3.10.A.9, is sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the COLD SHUTDOWN condition with all control rod assemblies inserted. This includes a 1% $\Delta k/k$ and a 50 ppm boron concentration allowance for uncertainty. This concentration is also sufficient to maintain the core subcritical with no control rod assemblies inserted into the reactor. Checks are performed during the reload design and safety analysis process to ensure the K-effective is equal to or less than 0.95 for each core. Periodic checks of refueling water boron concentration assure the proper shutdown margin. Specification 3.10.A.10 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.

In addition to the above safeguards, interlocks are used during refueling to assure safe handling of the fuel assemblies. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

Upon each completion of core loading and installation of the reactor vessel head, specific mechanical and electrical tests will be performed prior to initial criticality.

The fuel handling accident has been analyzed based on the methodology outlined in Regulatory Guide 1.183. The analysis assumes 100% release of the gap activity from the assembly with maximum gap activity after a 100-hour decay period following operation at 2605 MWt.

Detailed procedures and checks insure that fuel assemblies are loaded in the proper locations in the core. As an additional check, the movable incore detector system will be used to verify proper power distribution. This system is capable of revealing any assembly enrichment error or loading error which could cause power shapes to be peaked in excess of design value.

References

UFSAR Section 5.2	Containment Isolation
UFSAR Section 6.3	Consequence Limiting Safeguards
UFSAR Section 9.12	Fuel Handling System
UFSAR Section 11.3	Radiation Protection
UFSAR Section 13.3	Table 13.3-1
UFSAR Section 14.4.1	Fuel Handling Accidents
FSAR Supplement:	Volume I: Question 3.2

3.11 RADIOACTIVE GAS STORAGE

Applicability

Applies to the storage of radioactive gases.

Objective

To establish conditions by which gaseous waste containing radioactive materials may be stored.

Specification

A. Explosive Gas Mixture

1. The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration could exceed 4% by volume.

   a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

   b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the affected tank and reduce the concentration of oxygen to less than or equal to 4% by volume, then take the action in 1.a above.

   c. With the requirements of action 1.a above not satisfied, immediately suspend all additions of waste gases to the affected tank until the oxygen concentration is restored to less than or equal to 2% by volume, and submit a special report to the Commission within the next 30 days outlining the following:

      (1) The cause of the waste gas decay tank exceeding the 2% oxygen limit.

      (2) The reason why the oxygen concentration could not be returned to within limits.

(3) The actions taken and the time required to return the oxygen concentration to within limits.

2. The requirements of Specification 3.0.1 are not applicable.

B. Gas Storage Tanks

1. The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 24,600 curies of noble gases (considered as Xe-133).
2. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all addition of radioactive material to the tank and within 48 hours reduce the tank contents to within the limits.
3. The requirements of Specification 3.0.1 are not applicable.

Basis

Explosive Gas Mixture

Specification 3.11.A is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining oxygen below the concentration that will support combustion at any concentration of hydrogen 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 50.

Gas Storage Tanks

The tanks included in Specification 3.11.B are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that 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 in an event of 2 hours.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that 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. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, the shutdown control rod assemblies shall be fully withdrawn. With a shutdown control rod assembly not fully withdrawn, within 1 hour either fully withdraw the assembly or declare the assembly inoperable and apply Specification 3.12.C.

2. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, the full length control banks shall be inserted no further than the appropriate limit specified in the CORE OPERATING LIMITS REPORT. With a control bank inserted beyond the limit specified in the CORE OPERATING LIMITS REPORT, restore the control rod assembly bank to within its limits within 2 hours, or reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED POWER specified in the CORE OPERATING LIMITS REPORT, or place the reactor in HOT SHUTDOWN within 6 hours.

3. The Control Bank Insertion Limits shown in the CORE OPERATING LIMITS REPORT may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following:

a. The sequence of withdrawal of the control banks, when going from zero to 100% power, is A, B, C, D.

b. An overlap of control banks, consistent with physics calculations and physics data obtained during unit startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation 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 ($T_{avg} \geq 547°F$) 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.

4. Whenever the reactor is subcritical, except for physics tests, the critical control rod assembly position, i.e., the control rod assembly position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Insertion limits do not apply during physics tests or during periodic surveillance testing of control rod assemblies. However, the shutdown margin indicated above must be maintained except for the LOW POWER PHYSICS TEST to measure control and shutdown bank worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod assembly, expected to have the highest worth, inserted.

6. With a maximum of one control or shutdown bank inserted beyond the insertion limit specified in Specification 3.12.A.2 during control rod assembly testing pursuant to Specification 4.1, and immovable due to a failure of the Rod Control System, POWER OPERATION

may continue* provided that:

a. the affected bank insertion is limited to 18 steps below the insertion limit as measured by the group step counter demand position indicators,

b. the affected bank is trippable,

c. each control rod assembly is aligned to within ± 12 steps of its respective group step counter demand position indicator,

d. The shutdown margin requirement of Specification 3.12.A.3.c is determined to be met at least every 12 hours thereafter, and

e. the affected bank is restored to within the insertion limits of Specification 3.12.A within 72 hours.

Otherwise place the unit in HOT SHUTDOWN within the next 6 hours.

B. Power Distribution Limits

1. At all times except during LOW POWER PHYSICS TESTS, the hot channel factors defined in the basis meet the following limits:

$F_Q(Z) \leq (CFQ/P) \times K(Z)$ for $P > 0.5$
$F_Q(Z) \leq (CFQ/0.5) \times K(Z)$ for $P \leq 0.5$

where: CFQ = the FQ limit at RATED POWER specified in the CORE OPERATING LIMITS REPORT,

$$P = \frac{\text{THERMAL POWER}}{\text{RATED POWER}}$$, and

K(Z) = the normalized FQ limit as a function of core height, Z, as specified in the CORE OPERATING LIMITS REPORT

$F\Delta H(N) \leq CFDH \times (1 + PFDH \times (1-P))$

where: CFDH = the FΔH(N) limit at RATED POWER specified in the CORE OPERATING LIMITS REPORT,

PFDH = the Power Factor Multiplier for FΔH(N) specified in the CORE OPERATING LIMITS REPORT, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED POWER}}$$

* Provision for continued operation does not apply to Control Bank D inserted beyond the insertion limit.

2. Prior to exceeding 75% of RATED POWER following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor F_Q^{Meas} shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor $F_{\Delta H}^{N}$ shall be compared directly to the limit specified in Specification 3.12.B.1. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the FQ(Z) and $F_{\Delta H}^{N}$ limits as specified in the CORE OPERATING LIMITS REPORT within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced within the next 4 hours.

   b. The provisions of Specification 4.0.4 are not applicable.

3. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level P_O is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rod assemblies more than 190 steps withdrawn. The target flux difference at any other power level P is equal to the target value at P_O multiplied by the ratio P/P_O. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurements or by linear interpolation using the most recent value and the value predicted for the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

4. Except as modified by Specifications 3.12.B.4.a, b, c, or d below, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference (defines the target band on axial flux difference).

a. At a power level greater than 90 percent of RATED POWER, if the indicated axial flux difference deviates from its target band, within 15 minutes either restore the indicated axial flux difference to within the target band or reduce the reactor power to less than 90 percent of RATED POWER.

b. At a power level less than or equal to 90 percent of RATED POWER,

(1) The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference is within the limits shown on TS Figure 3.12-3. One minute penalty is accumulated for each one minute of operation outside of the target band at power levels equal to or above 50% of RATED POWER.

(2) If Specification 3.12.B.4.b.(1) is violated, then the reactor power shall be reduced to less than 50% power within 30 minutes and the high neutron flux setpoint shall be reduced to less than or equal to 55% power within the next four hours.

(3) A power increase to a level greater than 90 percent of RATED POWER is contingent upon the indicated axial flux difference being within its target band.

(4) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to TS Table 4.1-1 provided the indicated axial flux difference is maintained within the limits of TS Figure 3.12-3. A total of 16 hours of operation may be accumulated with the axial flux difference outside of the target band during this testing without penalty deviation.

c. At a power level less than or equal to 50 percent of RATED POWER,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50 percent of RATED POWER is contingent upon the indicated axial flux difference not being outside its target band for more than one hour accumulated penalty during the preceding 24-hour period. One half minute penalty is accumulated for each one minute of operation outside of the target band at power levels between 15% and 50% of RATED POWER.

d. The axial flux difference limits for Specifications 3.12.B.4.a, b, and c may be suspended during the performance of physics tests provided:

(1) The power level is maintained less than or equal to 85% of RATED POWER, and

(2) The limits of Specification 3.12.B.1 are maintained. The power level shall be determined to be less than or equal to 85% of RATED POWER at least once per hour during physics tests. Verification that the limits of Specification 3.12.B.1 are being met shall be demonstrated through in-core flux mapping at least once per 12 hours.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in Specification 3.12.B.4.a and the flux difference time limits in Specifications 3.12.B.4.b and c. If the alarms are out of service temporarily, the axial flux difference shall be logged and conformance to the limits assessed every hour for the first 24 hours and half-hourly thereafter. The indicated axial flux difference for each excore channel shall be monitored at least once per 7 days when the alarm is OPERABLE and at least once per hour for the first 24 hours after restoring the alarm to OPERABLE status.

5. The allowable QUADRANT POWER TILT is 2.0% and is only applicable while operating at THERMAL POWER > 50%.

6. If, except for operation at THERMAL POWER ≤ 50% or for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then:

   a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or

   b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

   c. If the QUADRANT POWER TILT exceeds ± 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

7. If, except for operation at THERMAL POWER ≤ 50% or for physics and control rod assembly surveillance testing, after a further period of 24 hours, the QUADRANT POWER TILT in Specification 3.12.B.5 above is not corrected to less than 2%:

   a. If the design hot channel factors for RATED POWER are not exceeded, an evaluation as to the cause of the discrepancy shall be made and a special report issued to the Nuclear Regulatory Commission.

   b. If the design hot channel factors for RATED POWER are exceeded and the power is greater than10%, then the high neutron flux, Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced 1% for each percent the hot channel factor exceeds the RATED POWER design values within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

c. If the hot channel factors are not determined, then the Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% power for every 1% QUADRANT POWER TILT within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

C. Control Rod Assemblies

1. To be considered OPERABLE during startup and POWER OPERATION each control rod assembly shall :
   1) be trippable,
   2) aligned within ± 24 steps of its group step demand position during the "Thermal Soak" period, as defined in Section 3.12.E.1.b, or ± 12 steps otherwise during power operation, and
   3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

2. To be considered OPERABLE during shutdown modes, each control rod assembly shall:
   1) be trippable,
   2) have its rod position indicator capable of verifying rod movement upon demand, and
   3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

3. Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:
   a. The control rod assembly is restored to OPERABLE status. as defined in Specification 3.12.C.1 and 2, or

   b. the shutdown margin requirement of Specification 3.12 A 3 c is satisfied. POWER OPERATION may then continue provided that:
      1) either:

(a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or

(b) the remainder of the control rod assemblies in the group with the inoperable control rod assembly are aligned to within 12 steps of the inoperable rod within one (1) hour while maintaining the control rod assembly sequence and insertion limits of Figure 3.12-1A and B; the THERMAL POWER level shall be restricted pursuant to Specification 3.12.A during subsequent operation.

2) the shutdown margin requirement of Specification 3.12.A.3.c is determined to be met within one hour and at least once per 12 hours thereafter.

3) the hot channel factors are shown to be within the design limits of Specification 3.12.B.1 within 72 hours. Further, it shall be demonstrated that the value of Fxy(Z) used in the Constant Axial Offset Control analysis is still valid.

4) a reevaluation of each accident analysis of Table 3.12-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions.

5) If power has been reduced in accordance with Specification 3.12.C.3.b, power may be increased above 75% of RATED POWER provided that:

(a) an analysis has been performed to determine the hot channel factors and the resulting allowable power level based on the limits of Specification 3.12.B.1, and

(b) an evaluation of the effects of operating at the increased power level on the accident analyses of Table 3.12-1 has been completed.

4. With more than one inoperable control rod assembly, as defined in Specification 3.12.C.1, determine within 1 hour that the shutdown margin requirement of Specification 3.12.A.3.c is satisfied and be in HOT SHUTDOWN within 6 hours.

5. The provisions of Specifications 3.12.C.1 and 3.12.C.4 shall not apply during LOW POWER PHYSICS TESTS in which the control rod assemblies are intentionally misaligned.

D. QUADRANT POWER TILT

1. If the reactor is operating above 75% of RATED POWER with one excore nuclear channel out of service, the QUADRANT POWER TILT shall be determined:
   a. Once per day, and
   b. After a change in power level greater than 10% or more than 30 inches of control rod motion.

2. The QUADRANT POWER TILT shall be determined by one of the following methods:
   a. Movable detectors (at least two per quadrant)
   b. Core exit thermocouples (at least four per quadrant)

E. Rod Position Indication System

1. Rod position indication shall be provided as follows:

   a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

   b. From movement of control banks to achieve criticality up to 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 24 steps of their respective group step demand counter indications for a maximum of one hour out of twenty-four, and to within ± 12 steps otherwise. During the one-hour "Thermal Soak" period, the step demand counters shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps.

   c. In HOT, INTERMEDIATE, and COLD SHUTDOWN, the step demand counters shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps. The rod position indicators shall be available to verify control rod assembly movement upon demand.

2. If a rod position indicator channel is inoperable, then:

   a. For operation above 50% of RATED POWER, the position of the control rod assembly shall be checked indirectly using the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating control rod assembly exceeding 24 steps, or

   b. Reduce power to less than 50% of RATED POWER within 8 hours. During operations below 50% of RATED POWER, no special monitoring is required.

3. If more than one rod position indicator channel per group or two rod position indicator channels per bank are inoperable during control bank motion to achieve criticality or POWER OPERATION, then the unit shall be placed in HOT SHUTDOWN within 6 hours.

F. DNB Parameters

1. The following DNB related parameters shall be maintained within their limits during POWER OPERATION:

- Reactor Coolant System $T_{avg} \leq 577.0°F$
- Pressurizer Pressure ≥ 2205 psig
- Reactor Coolant System Total Flow Rate ≥ 273,000 gpm

a. The Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED POWER within the next 4 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED POWER per minute or a THERMAL POWER step increase in excess of 10% of RATED POWER.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to COLD SHUTDOWN) are compensated for by changes in the soluble boron concentration. During POWER OPERATION, the shutdown control rod assemblies are fully withdrawn and control of power is by the control banks. A reactor trip occurring during POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement.

The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analyses of the hypothetical steam break accident. The control rod assembly insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length is established at 1.77% reactivity. Other accident analyses with the exception of the Chemical and Volume Control System malfunction analyses are based on 1% reactivity shutdown margin. Relative positions of control banks are determined by a specified control bank overlap. This overlap is based on the consideration of axial power shape control. The specified control rod assembly insertion limits have been established to limit the potential ejected control rod assembly worth in order to account for the effects of fuel densification. The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank, that is, with each assembly in the bank within one step (5/8 inch) of the bank position.

Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks, and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Variable Differential Transformer is approximately ±5% of span (±12 steps) under steady state conditions. The relative accuracy of the linear position indicator has been considered in establishing the maximum allowable deviation of a control rod assembly from its indicated group step demand position. In the event that the linear position indicator is not in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (full length control rod assembly 12 feet out of alignment with its bank), operation at 50% steady state power does not result in exceeding core limits.

The "Thermal Soak" allowance below 50% power, during which the Rod Position Indication System tolerance requirement is relaxed, provides time for the system to reach thermal equilibrium. A total of one hour in twenty-four is available for this allowance, which may be a continuous hour or may consist of discrete, shorter intervals. For such a short period of time, a misaligned control rod assembly does not pose an unacceptable risk. At these conditions, the rod position indicators should still be used to verify rod movement but not their exact location. The tolerance is tightened after one hour to ensure that the thermal overshoot does not conceal an actual control rod assembly misalignment.

The reliance upon the step demand counters at HOT and COLD SHUTDOWN shifts the monitoring of control rod assembly position from the Rod Position Indication System to the more reliable demand counters when Reactor Coolant System temperature is changing greatly but the core remains subcritical. The step demand counters also provide precise group demand positions during the thermal soak period.

The specified control rod assembly drop time is consistent with safety analyses that have been performed.

An inoperable control rod assembly imposes additional demands on the operators. The permissible number of inoperable control rod assemblies is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the OPERABLE control rod assemblies upon reactor trip.

In the event that a failure of the Rod Control System renders control rod assemblies immovable, provision is made for continued operation provided:

- the affected control rod assemblies remain trippable,
- the individual control rod assembly alignment limits are met.

In the event that a failure of the Rod Control System renders control rod assembly banks immovable during control rod assembly surveillance testing, provision is made for 72 hours of continued operation provided:

- the affected control rod assemblies remain trippable,
- the individual control rod assembly alignment limits are met,
- a maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit, and
- the shutdown margin requirements are verified every 12 hours during the period the insertion limit is not met.

The 72 hour provision does not apply to Control Bank D since insertion of D bank below the insertion limit is not required for control rod assembly surveillance testing.

Checks are performed for each reload core to ensure that this minor bank insertion will not result in power distributions which violate the Departure from Nucleate Boiling (DNB) criterion for ANS Condition II transient (moderate frequency transients analyzed in Section 14.2 of the UFSAR) during the repair period or in a violation of the shutdown margin requirements of Specification of 3.12.A.3.c during the repair period.

The 72 hour period for a control rod assembly bank to be inserted below its limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of fuel centerline temperature must not exceed 4700°F. Second, the minimum DNB Ratio (DNBR) in the core must not be less than the applicable design limit in normal operation or in short term transients.

In addition to the above, the peak linear power density and the nuclear enthalpy rise hot channel factor must not exceed their limiting values which result from the large break loss of coolant accident analysis based on the Emergency Core Cooling System acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits of power distribution, the following hot channel factors are defined:

> $F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerance on fuel pellets and rods.
>
> 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 for non-statistical applications.
>
> $F_{\Delta H}^N$, 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 for both loss of coolant accident and non-loss of coolant accident considerations.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and loss of coolant accident calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the Emergency Core Cooling System acceptance criteria as specified in 10 CFR 50.46 using the upper bound $F_Q(Z)$ times the hot channel factor normalized operating envelope given in the CORE OPERATING LIMITS REPORT.

When an F_Q measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowance for measurement error for a full core map (greater than or equal to 38 thimbles, including a minimum of 2 thimbles per core quadrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of F_Q.

In the $F^N_{\Delta H}$ limit specified in the CORE OPERATING LIMITS REPORT, there is a four percent error allowance, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq$ CFDH [1 + PFDH (1-P)]/1.04. The 4% allowance is based on the considerations that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence F_Q, can be compensated for by tighter axial control. An appropriate allowance for the measurement uncertainty for $F^N_{\Delta H}$ obtained from a full core map (≥ 38 thimbles, including a minimum of 2 detectors per core quadrant, monitored) taken with the movable incore detector flux mapping system has been incorporated in the statistical DNBR limit.

Measurement of the hot channel factors are required as part of startup physics tests, during 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 core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic 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 has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor $F^N_{\Delta H}$ limit will be met. These conditions are as follows:

1. Control rod assemblies in a single bank move together with no individual control rod assembly insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a control rod assembly misalignment no greater than 15 inches with consideration of maximum instrumentation error.
2. Control rod banks are sequenced with overlapping banks as shown in the Control Bank Insertion Limits specified in the CORE OPERATING LIMITS REPORT.

3. The full length Control Bank Insertion Limits specified in the CORE OPERATING LIMITS REPORT 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 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 the bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ with decreasing power level allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, this hot channel factor limit is met.

A recent evaluation of DNB test data obtained from experiments of fuel rod bowing in thimble cells has identified that the reduction in DNBR due to rod bowing in thimble cells is more than completely accommodated by existing thermal margins in the core design. Therefore, it is not necessary to continue to apply a rod bow penalty to $F^N_{\Delta H}$.

The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control given in Specification 3.12.B.4 together with the surveillance requirements given in Specification 3.12.B.2 assure that the Limiting Condition for Operation for the heat flux hot channel factor is met.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating, is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5% ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not always possible during certain physics tests or during excore detector calibrations. Therefore, the specifications on power distribution control are less restrictive during physics tests and excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid unit 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 instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range ±13.8 percent (±10.8 percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

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 rod assemblies to produce the required indicated flux difference.

A 2% QUADRANT POWER TILT allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod assembly and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rod assemblies producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The QPTR limit must be maintained during power operation with THERMAL POWER > 50% of RATED POWER to prevent core power distributions from exceeding the design limits.

Applicability during power operation ≤ 50% RATED POWER or when shut down is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RATED POWER or lower.

The limits of the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The measurement of the Reactor Coolant System Total Flow Rate once per refueling cycle is adequate to detect flow degradation.

TABLE 3.12-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE CONTROL ROD ASSEMBLY

Control Rod Assembly Insertion Characteristics

Control Rod Assembly Misalignment

Large and Small Break Loss of Coolant Accidents

Single Reactor Coolant Pump Locked Rotor

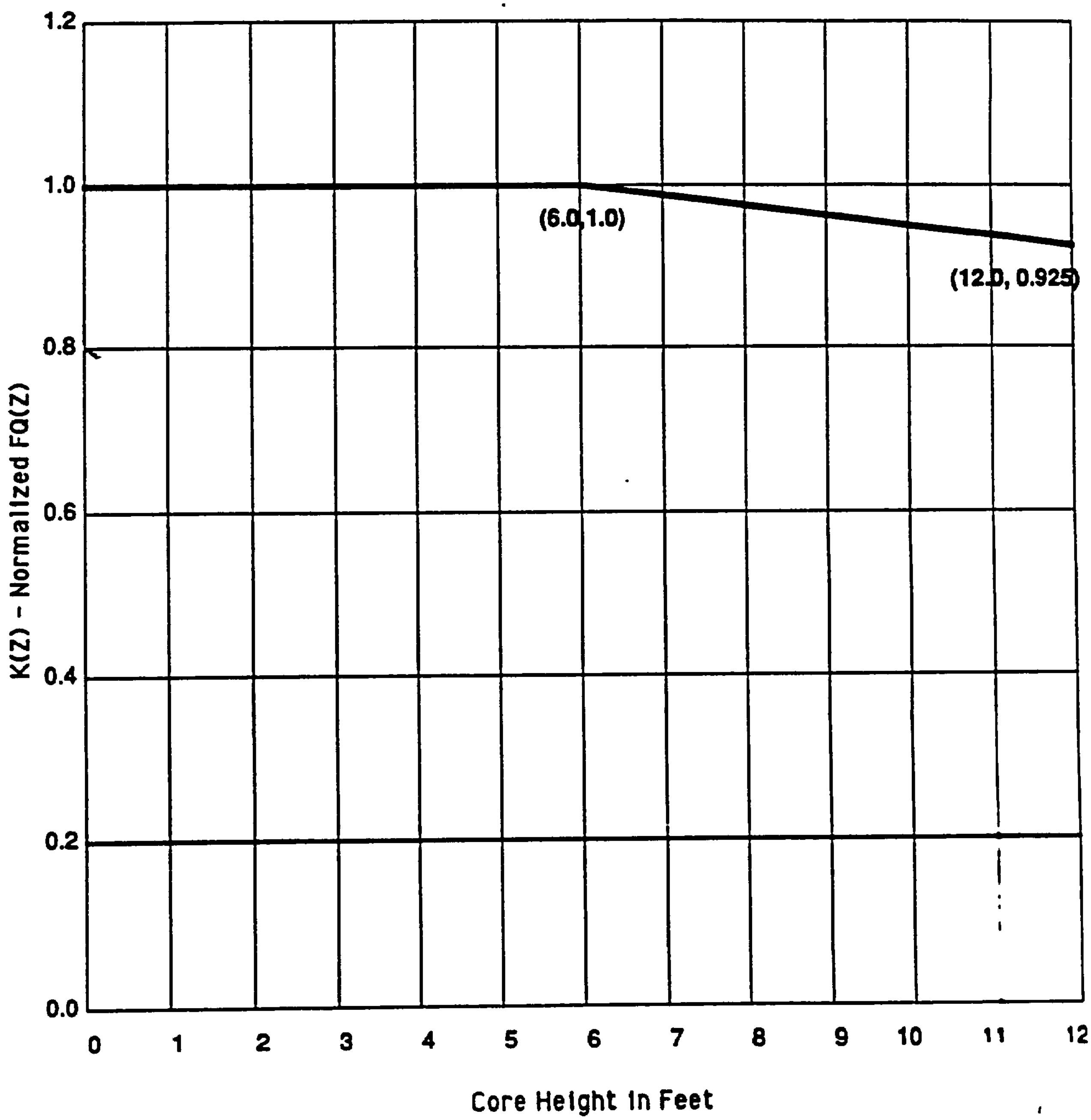
Major Secondary Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing
(Control Rod Assembly Ejection)

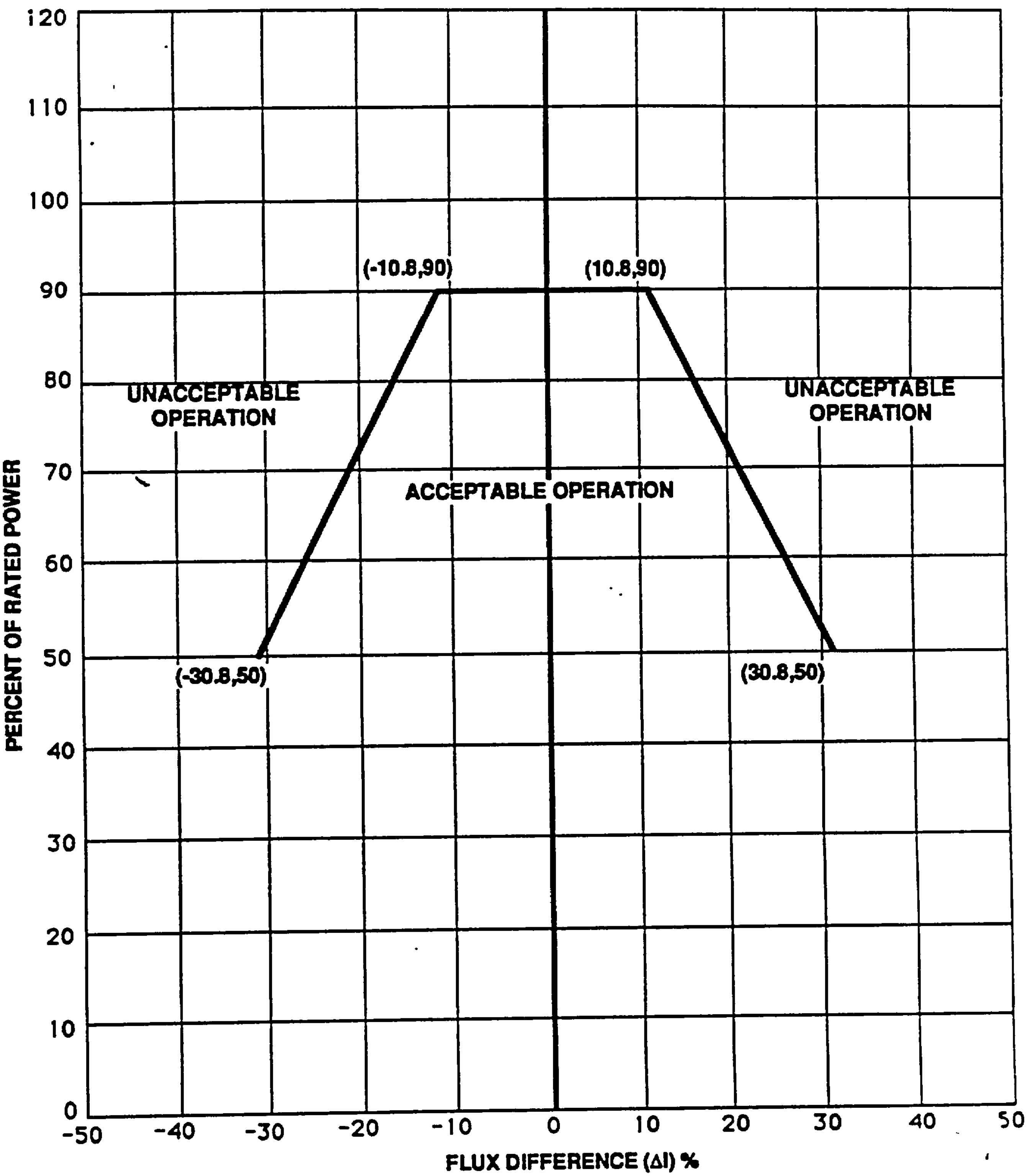
DELETED

DELETED

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



AXIAL FLUX DIFFERENCE LIMITS
AS A FUNCTION OF RATED POWER
SURRY POWER STATION



3.13 COMPONENT COOLING SYSTEM

Applicability

Applies to the operational status of all subsystems of the Component Cooling System. The Component Cooling System consists of the Component Cooling Water Subsystem, Chilled Component Water Subsystem, Chilled Water Subsystem, and Neutron Shield Tank Cooling Water Subsystem.

Objective

To define limiting conditions for each subsystem of the Component Cooling System necessary to assure safe operation of each reactor unit of the station during startup, POWER OPERATION, or cooldown.

Specifications

A. When a unit's Reactor Coolant System temperature and pressure exceed 350°F and 450 psig, respectively, or when a unit's reactor is critical operating conditions for the Component Cooling Water Subsystem shall be as follows:

   1. For one unit operation, two component cooling water pumps and heat exchangers shall be OPERABLE.

   2. For two unit operation, three component cooling water pumps and heat exchangers shall be OPERABLE.

   3. The Component Cooling Water Subsystem shall be OPERABLE for immediate supply of cooling water to the following components, if required:

      a. Two OPERABLE residual heat removal heat exchangers.

B. During POWER OPERATION, Specification A-1, A-2, or A-3 above may be modified to allow one of the required components to be inoperable provided immediate attention is directed to making repairs. If the system is not restored within 24 hours to the requirements of Specification A-1, A-2, or A-3, an operating reactor shall be placed in HOT SHUTDOWN within the next 6 hours. If the repairs are not completed within an additional 48 hours, the affected reactor shall be placed in COLD SHUTDOWN within the following 30 hours.

C. Whenever the component cooling water radiation monitor is inoperable, the surge tank vent valve shall remain closed.

Basis

The Component Cooling System is an intermediate cooling system which serves both reactor units. It transfers heat from heat exchangers containing reactor coolant, other radioactive liquids, and other fluids to the Service Water System. The Component Cooling System is designed to (1) provide cooling water for the removal of residual and sensible heat from the Reactor Coolant System during shutdown, cooldown, and startup, (2) cool the containment recirculation air coolers and the reactor coolant pump motor coolers, (3) cool the letdown flow in the Chemical and Volume Control System during POWER OPERATION, and during residual heat removal for continued purification, (4) cool the reactor coolant pump seal water return flow, (5) provide cooling water for the neutron shield tank and (6) provide cooling to dissipate heat from other reactor unit components.

The Component Cooling Water Subsystem has four component cooling water pumps and four component cooling water heat exchangers. Each of the component cooling water heat exchangers is designed to remove during normal operation the entire heat load from one unit plus one half of the heat load common to both units. Thus, one component cooling water pump and one component cooling water heat exchanger are required for each unit which is at POWER OPERATION. Two pumps and two heat exchangers are normally operated during the removal of residual and sensible heat from one unit during cooldown. Failure of a single component may extend the time required for cooldown but does not affect the safe operation of the station.

References

UFSAR Section 5.3, Containment Systems
UFSAR Section 9.4, Component Cooling System
UFSAR Section 15.5.1.2, Containment Design Criteria

3.14 CIRCULATING AND SERVICE WATER SYSTEMS

Applicability

Applies to the operational status of the Circulating and Service Water Systems.

Objective

To define those limiting conditions of the Circulating and Service Water Systems necessary to assure safe station operation.

Specification

A. The Reactor Coolant System temperature or pressure of a reactor unit shall not exceed 350° F or 450 psig, respectively, or the reactor shall not be critical unless:

1. The high level intake canal is filled to at least elevation −23.0 feet at the high level intake structure.

2. Unit subsystems, including piping and valves, shall be operable to the extent of being able to establish the following:

   a. Flow to and from one bearing cooling water heat exchanger.

   b. Flow to and from the component cooling heat exchangers required by Specification 3.13.

3. At least two circulating water pumps are operating or are operable.

4. Three emergency service water pumps are operable; these pumps will service both units simultaneously.

5. Two service water flow paths to the charging pump service water subsystem are OPERABLE.

6. Two service water flow paths to the recirculation spray subsystems are OPERABLE.

B. The requirements of Specification 3.14.A.4 may be modified to allow one Emergency Service Water pump to remain inoperable for a period not to exceed 7 days. If this pump is not OPERABLE in 7 days, then place both units in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the next 30 hours.

The requirements of 3.14.A.4 may be modified to have two Emergency Service Water pumps OPERABLE with one unit in COLD SHUTDOWN with combined Spent Fuel pit and shutdown unit decay heat loads of 25 million BTU/HR or less. One of the two remaining pumps may be inoperable for a period not to exceed 7 days. If this pump is not OPERABLE in 7 days, then place the operating unit in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the next 30 hours.

C. There shall be an operating service water flow path to and from one operating main control and emergency switchgear rooms air conditioning condenser and at least one OPERABLE service water flow path to and from at least one OPERABLE main control and emergency switchgear rooms air conditioning condenser whenever fuel is loaded in the reactor core. Refer to Section 3.23.C for air conditioning system operability requirements above COLD SHUTDOWN.

D. The requirements of Specifications 3.14.A.5, 3.14.A.6, and 3.14.C may be modified to allow unit operation with only one OPERABLE flow path to the charging pump service water subsystem, the recirculation spray subsystems, and to the main control and emergency switchgear rooms air conditioning condensers. If the affected systems are not restored to the requirements of Specifications 3.14.A.5, 3.14.A.6, and 3.14.C within 24 hours, the reactor shall be placed in HOT SHUTDOWN. If the requirements of Specifications 3.14.A.5, 3.14.A.6, and 3.14.C are not met within an additional 48 hours, the reactor shall be placed in COLD SHUTDOWN.

Basis

The Circulating and Service Water Systems are designed for the removal of heat resulting from the operation of various systems and components of either or both of the units. Untreated water, supplied from the James River and stored in the high level intake canal is circulated by gravity through the recirculation spray coolers and the bearing cooling water heat exchangers and to the charging pumps lubricating oil cooler service water pumps which supply service water to the charging pump lube oil coolers.

In addition, the Circulating and Service Water Systems supply cooling water to the component cooling water heat exchangers and to the main control and emergency switchgear rooms air conditioning condensers. The Component Cooling heat exchangers are used during normal plant operations to cool various station components and when in shutdown to remove residual heat from the reactor. Component Cooling is not required on the accident unit during a loss-of-coolant accident. If the loss-of-coolant accident is coincident with a loss of off-site power, the nonaccident unit will be maintained at HOT SHUTDOWN with the ability to reach COLD SHUTDOWN.

The long term Service Water requirement for a loss-of-coolant accident in one unit with simultaneous loss-of-station power and the second unit being brought to HOT SHUTDOWN is greater than 15,000 gpm. Additional Service Water is necessary to bring the nonaccident unit to COLD SHUTDOWN. Three diesel driven Emergency Service Water pumps with a design capacity of 15,000 gpm each, are provided to supply water to the High Level Intake canal during a loss-of-station power incident. Thus, considering the single active failure of one pump, three Emergency Service Water pumps are required to be OPERABLE. The allowed outage time of 7 days provides operational flexibility to allow for repairs up to and including replacement of an Emergency Service Water pump without forcing dual unit outages, yet limits the amount of operating time without the specified number of pumps

When one Unit is in Cold Shutdown and the heat load from the shutdown unit and spent fuel pool drops to less than 25 million BTU HR, then one Emergency Service Water pump may be removed from service for the subsequent time that the unit remains in Cold Shutdown due to the reduced residual heat removal and hence component cooling requirements

A minimum level of −17.2 feet in the High Level Intake canal is required to provide design flow of Service Water through the Recirculation Spray heat exchangers during a loss-of-coolant accident for the first 24 hours. If the water level falls below −23' 6", signals are generated to trip both unit's turbines and to close the nonessential Circulating and Service Water valves. A High Level Intake canal level of −23' 6" ensures actuation prior to canal level falling to elevation −23' The Circulating Water and Service Water isolation valves which are required to close to conserve Intake Canal inventory are periodically verified to limit total leakage flow out of the Intake Canal In addition, passive vacuum breakers are installed on the Circulating Water pump discharge lines to assure that a reverse siphon is not continued for canal levels less than −23 feet when Circulating Water pumps are de-energized The remaining six feet of canal level is provided coincident with ESW pump operation as the required source of Service Water for heat loads following the Design Basis Accident

References

UFSAR Section 9.9	Service Water System
UFSAR Section 10.3.4	Circulating Water System
UFSAR Section 14.5	Loss-of-Coolant Accidents, Including the Design Basis Accident