50-346

Davis Besse Unit 1 Technical Specifications with Previously Accepted Changes.

w/ltr DTD. 8-13-76

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to detect and monitor leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that, while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED LEAKAGE portion of this can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limit of (10) GPM restricts operation with a total RCS leakage to all RC pump seals in excess of ( ) GPM.

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The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curves, Figures 3.4-2 and 3.4-3, are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour. The cooldown limit curves, Figures 3.4-2 and 3.4-3 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 heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 2 5 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in BASES Table 4-1. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using BASES Figures 4-1 and 4-2. The heatup and cooldown limit curves, of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 2 5 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be

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BASES

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems with RCS average temperature > 280°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the core flooding tanks is capable of supplying sufficient core cooling to maintain the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures, that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. ~~Power is required to be removed from any valve which fails to meet single failure criteria~~. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between (8.5) and (11.0) of the solution recirculated within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

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3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psi and 2) the containment peak pressure does not exceed the design pressure of 40 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 37 psig. The limit of 1 psig for initial positive containment pressure will limit the total pressure to 38 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of ~~(40)~~ 38 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment

BASES

3/4.6.5 COMBUSTIBLE GAS CONTROL

~~The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.~~

See previous basis

3/4.6.8 SECONDARY CONTAINMENT

3/4.6.8.1 EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the emergency ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

BASES

3/4.6.5 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the Hydrogen Analyzers, Containment Hydrogen Dilution System, Containment Recirculation System and Hydrogen Purge System ensures that this equipment will be available to maintain the maximum hydrogen concentration within the containment vessel at or below three volume percent following a LOCA.

The two redundant Hydrogen Analyzers determine the content of hydrogen within the containment vessel.

The Containment Hydrogen Dilution (CHD) System consists of two full capacity, redundant, rotary, positive displacement type blowers to supply air to the containment. The CHD System controls the hydrogen concentration by the addition of air to the containment vessel, resulting in a pressurization of the containment and suppression of the hydrogen volume fraction.

The Containment Recirculation System is designed to draw from the areas of potentially high hydrogen concentrations in the containment dome and provide a more uniform dispersion of hydrogen throughout the containment vessel. The system consists of two redundant fans with independent duct distribution systems.

The Containment Hydrogen Purge System Filter Unit functions as a backup to the CHD System and is designed to release air from the containment atmosphere through a HEPA filter and charcoal filter prior to discharge to the station vent.

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PLANT SYSTEMS

BASES

3/4 [illegible] 2 AUXILIARY FEEDWATER SYSTEMS

The OPERABILITY of the auxiliary feedwater systems ensures that the Reactor Coolant System can be cooled down to less than 280°F from normal operating conditions in the event of a total loss of offsite power.

Each steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of ~~700~~ 850 gpm at a pressure of ~~1133~~ 1035 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than ~~[illegible]°F~~ 280°F where the Decay Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE FACILITIES

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 280°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the

3/4.10.1 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

withdrawal

This special test exception permits individual control rods to be positioned outside of their specified group heights and ~~insertion~~ limits and to be assigned to other than specified control rod groups, and permits AXIAL POWER IMBALANCE and QUADRANT POWER TILT limits to be exceeded during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth, 2) determine the reactor stability index and damping factor under xenon oscillation conditions and 3) calibrate AXIAL POWER IMBALANCE and QUADRANT POWER TILT instrumentation.

3/4.10.2 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation.

3/4.10.3 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required in order to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

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EXCLUSION AREA

FIGURE 5.1-1

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LOW POPULATION ZONE

FIGURE 5.1-2

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,110 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 with the storage pool filled with unborated water. The K_{eff} of ≤ 0.95 includes a conservative allowance of 1% $\Delta k/k$ for uncertainties as described in Section ~~9.1~~ of the FSAR.

2.1

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ________.

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CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to more than 260 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1.

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OFFSITE ORGANIZATION

Figure 6.2-1

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FACILITY ORGANIZATION

Figure 6.2-2

6.3 STATION STAFF QUALIFICATIONS

6.3.1 Each member of the station staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, ~~except for the Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the station staff shall be maintained under the direction of the Training Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 STATION REVIEW BOARD (SRB)

FUNCTION

6.5.1.1 The Station Review Board (SRB) shall function to advise the Station Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.2.2 The Company Nuclear Review Board shall be composed of the:

Chairman:	**Vice President, Facilities Development**
Member:	Vice President, Energy Supply
Member:	General Superintendent, Power Engineering and Construction
Member:	General Superintendent, Transmission and Substations
Member:	Superintendent, Davis-Besse Station
Member:	Superintendent, Heavy Maintenance
Member:	Superintendent, ~~Technical Services~~ Energy Services
Member:	Nuclear Engineer, Power Engineering
Member:	Manager, Quality Assurance

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of CNRB shall consist of the Chairman or his designated alternate and five of the CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of ≥ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510-1975 (130°C, 95% R.H.) and the samples prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

   a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3300 cfm ± 10%, and

   b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ____ cfm ± 10%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < ~~(6)~~ 4.5 inches Water Gauge while operating the ventilation system at a flow rate of 3300 cfm ± 10 percent.

   2. Verifying that the control room normal ventilation system is isolated by a SFAS test signal, Control Room Ventilation Air Intake Chlorine Concentration - High test signal, and a Station Vent Radiation High test signal.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the system maintains the control room at a positive pressure of ≥ 0.25 inches W.G. relative to the outside atmosphere at a system flow rate of 3300 cfm ± 10%.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are test in-place in accordance with ANSI N510-1975 while operating the filter system at a flow rate of 3300 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filter system at a flow rate of 3300 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

acceptable

4.7.7.1.3 At least once per 18 months ~~during shutdown~~ a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-4, whichever is less, shall be selected and functionally tested to verify ~~correct~~ piston movement, lock up and bleed. Snubbers greater than 50,000 lbs capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified in Table 3.7-4 as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. ~~Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation.~~ For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

functional

TABLE 4.7-4

SAFETY RELATED HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
≥8	31 days ± 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 250/125-volt D.C. MCC, and

2 - 125-volt battery banks and chargers supplying the above D.C. MCC.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 250/125-volt D.C. MCC shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery ~~bank~~ and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

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BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a Keff of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of $\geq$ ~~(1500)~~ 1800 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 10 gpm of 8750 ppm boric acid solution or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ ~~(1500)~~ 1800 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any safety or regulating rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor pressure vessel and the refueling canal shall be determined by chemical analysis at least once each 72 hours.

*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room, ~~and one with audible indication in the containment and control room~~.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

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CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment purge and exhaust isolation valve. as listed in table 3.6-2

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge and exhaust valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment by:

a. Verifying the penetrations are in their isolated condition, or

b. Testing the containment purge and exhaust valves per the applicable portions of Specification 4.6.3.1.2.

FUEL HANDLING BRIDGE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The fuel handling bridges shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. A hoist minimum capacity of 3250 pounds, and

b. A hoist overload cutoff limit of $\leq$ 2750 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for bridge OPERABILITY not satisfied, suspend use of any inoperable bridge from operations involving the movement of control rods or fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 Each fuel handling bridge used for movement of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of moving control rods or fuel assemblies by performing a hoist load test of at least 3250 pounds and demonstrating an automatic load cutoff when the hoist load exceeds 2750 pounds.

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CRANE TRAVEL - FUEL HANDLING BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of ___ pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies and water in the ~~storage~~ spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and/or physical stops which prevent crane travel with loads in excess of ____ pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane operation and at least once per 7 days during crane operation.

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COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one decay heat removal loop shall be ~~in operation~~. operable

APPLICABILITY: MODE 6.

ACTION:

a. With less than one decay heat removal loop in operation, except as provided in b below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The decay heat removal loop may be removed from operation for up to one hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel (hot) legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 At least one decay heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 2800 gpm at least once per 24 hours.

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge and exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the containment purge and exhaust isolation system inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge and exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge and exhaust isolation occurs on manual initiation and on a high radiation test signal from ~~each of~~ the SFAS ~~instrumentation channels.~~

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3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the ~~insertion~~ withdrawal limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, ~~and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident a minimum SHUTDOWN MARGIN of (0.30)% Δk/k is initially required to control the reactivity transient~~. Accordingly, the SHUTDOWN MARGIN required is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2800 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2800 GPM will circulate an equivalent Reactor Coolant System volume of 12,110 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

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3/4.1.2 BORATION SYSTEMS (Continued)

stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either ( ) gallons of (12,250) ppm borated water from the boric accid storage system or ( ) gallons of (1800) ppm borated water from the borated water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between (8.5) and (11.0) of the solution recirculated within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and ~~insertion~~ withdrawal limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

3/4.1.2 BORATION SYSTEMS (Continued)

stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

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The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.32 during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.4 kw/ft during normal operation, and (c) maintaining the peak power density $\leq$ 20.4 kw/ft during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power-imbalance envelope defined in Figures 3.2-1 and 3.2-2, and the insertion limit curves, Figures 3.1-1 and 3.1-3 are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power-imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power-imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined by Figures 3.1-1 and 3.1-3 and if a 4 percent QUADRANT POWER TILT exists. Additional conservatism is introducted by application of:

4.92

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$$F_Q \leq 2.94; \quad F^N_{\Delta H} \leq 1.71$$

Power Peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 6.5% (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained between + ~~14~~ 14.3 percent and - ~~20~~ 20.4 percent at RATED THERMAL POWER.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:

a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

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BASES

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding 20.4 kw/ft locally, and from going below a minimum DNBR of 1.32, by automatic protection on power, AXIAL POWER IMBALANCE, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the Reactor Protection System.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

4.92

The QUADRANT POWER TILT limit of 4% at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 4.5% can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 4% was selected to provide an allowance for the uncertainty associated with the indicated power tilt. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of 4%.

4.92 4.92

3/4.2.5 DNB PARAMETERS

The limits on 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 FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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INCORE INSTRUMENTATION SPECIFICATION
ACCEPTABLE MINIMUM AXIAL IMBALANCE ARRANGEMENT

BASES FIGURE 3-1

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INCORE INSTRUMENTATION SPECIFICATION
ACCEPTABLE MINIMUM RADIAL TILT ARRANGEMENT

BASES FIGURE 3-2

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.32 during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one or both loops, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE and the Nuclear Overpower Based on Pump Monitors trip, ensuring that the DNBR will be maintained above 1.30 at the maximum possible THERMAL POWER for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR equal to 22%, whichever is more restrictive.

1.32

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a DHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 336,000 lbs per hour of saturated steam at the valve's setpoint.

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating DHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent ~~a pressurizer low level~~ or a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the ~~Engineered~~ Safety Feature Actuation System as a result of a reactor trip ~~scram.~~ The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area Emergency Ventilation System Actuation	(1)	**	$\leq$(2) x background	(1 - 10^5) cpm	15
2. PROCESS MONITORS					
a. Fuel Storage Pool Area					
i. Gaseous Activity - Normal Ventilation System Isolation	1	*	$\leq 3 \times 10^{-9}$ µci/cc	10 - 10^6 cpm	15
ii. Particulate Activity - Normal Ventilation System Isolation	1	*	$\leq 1 \times 10^{-10}$ µci/cc	10 - 10^6 cpm	15
b. Containment					
i. Gaseous Activity					
~~a) Purge & Exhaust~~ Isolation	~~1~~	~~6~~	~~$\leq 3 \times 10^{-7}$ µci/cc~~	~~10 - 10^6 cpm~~	16
b) RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	10 - 10^6 cpm	16
ii. Particulate Activity					
~~a) Purge & Exhaust Isolation~~	~~1~~	~~6~~	~~$\leq 1 \times 10^{-10}$ µci/cc~~	~~10 - 10^6 cpm~~	16
b) RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	10 - 10^6 cpm	14

Covered by SFAS

*With irradiated fuel in the storage pool
**With fuel in the storage pool or building

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TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

~~ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.~~

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area Emergency Ventilation System Actuation	S	R	M	**
2. PROCESS MONITORS				
a. Fuel Storage Pool Area				
i. Gaseous Activity - Normal Ventilation System Isolation	S	R	M	*
ii. Particulate Activity - Normal Ventilation System Isolation	S	R	M	*
b. Containment				
i. Gaseous Activity				
~~a) Purge & Exhaust Isolation~~	~~S~~	~~R~~	~~M~~	~~6~~
b) RCS leakage detection	S	R	M	1, 2, 3 & 4
ii. Particulate Activity				
~~a) Purge & Exhaust Isolation~~	~~S~~	~~R~~	~~M~~	~~6~~
b) RCS Leakage Detection	S	R	M	1, 2, 3 & 4

*With irradiated fuel in the storage pool

**With irradiated fuel in the storage pool or building

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 As a minimum, the incore detectors shall be OPERABLE as specified below.

a. For AXIAL POWER IMBALANCE measurements:

1. ~~Nine~~ three detectors, ~~three~~ one in each of 3 strings, shall lie in the same axial plane with 1 plane in each axial core half.

2. The axial planes in each core half shall be symmetrical about the core mid-plane.

3. The detector strings shall not have radial symmetry.

b. For QUADRANT POWER TILT measurements:

1. Two sets of 4 detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

2. Detectors in the same plane shall have quarter core radial symmetry.

APPLICABILITY: When the incore detection system is used for surveillance of:

a. The AXIAL POWER IMBALANCE, or

b. The QUADRANT POWER TILT.

ACTION:

With less than the specified minimum incore detector arrangement OPERABLE, do not use incore detector measurements to determine AXIAL POWER IMBALANCE or QUADRANT POWER TILT. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detector system shall be demonstrated OPERABLE:

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENT OPERABLE
1. Triaxial Time-History Accelographs		
a. ________	________	1
b. ________	________	1
c. ________	________	1
d. ________	________	1
2. Triaxial Peak Accelographs		
a. ________	________	1
b. ________	________	1
c. ________	________	1
d. ________	________	1
e. ________	________	1
3. Triaxial Seismic Switches		
a. ________	________	1*
b. ________	________	1*
c. ________	________	1*
d. ________	________	1*
4. Triaxial Response-Spectrum Recorders		
a. ________	________	1*
b. ________	________	1
c. ________	________	1
d. ________	________	1
e. ________	________	1
f. ________	________	1

* With reactor control room indication

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. ______	M*	R	SA
b. ______	M*	R	SA
c. ______	M*	R	SA
d. ______	M*	R	SA
2. Triaxial Peak Accelographs			
a. ______	NA	R	NA
b. ______	NA	R	NA
c. ______	NA	R	NA
d. ______	NA	R	NA
e. ______	NA	R	NA
3. Triaxial Seismic Switches			
a. ______	**M	R	SA
b. ______	**M	R	SA
c. ______	**M	R	SA
d. ______	**M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. ______	**M	R	SA
b. ______	NA	R	SA
c. ______	NA	R	SA
d. ______	NA	R	SA
e. ______	NA	R	SA
f. ______	NA	R	SA

* Except seismic trigger

** With reactor control room indication

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM OPERABLE
1. WIND SPEED			
a. Nominal Elev.	612	± 0.5 mph*	1
b. Nominal Elev.	827	± 0.5 mph*	1
2. WIND DIRECTION			
a. Nominal Elev.	612	± 5°	1
b. Nominal Elev.	827	± 5°	1
3. AIR TEMPERATURE - DELTA T			
a. Nominal Elev.	~~612~~ 827-612	± 0.1°C	1
~~b. Nominal Elev.~~	~~827~~	~~± 0.1°C~~	~~1~~

*Starting speed of anemometer shall be < 1 mph.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMETATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 612	D	SA
b. Nominal Elev. 827	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 612	D	SA
b. Nominal Elev. 827	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 827-612	D	SA
~~b. Nominal Elev. 827~~	~~D~~	~~SA~~

See previously accepted spec which reflects DB equip.

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of $\leq$ 5 ppm, shall be OPERABLE.

APPLICABILITY: 1, 2, 3 and 4

ACTION:

a. With less than two chlorine detection systems OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation; restore the inoperable detection system to OPERABLE status within 30 day or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

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CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.6 The chlorine detection system, with the alarm/trip setpoint adjusted to actuate at a chlorine concentration of ≤ 5 ppm, shall be OPERABLE with, as a minimum, (two) OPERABLE chlorine detectors located in the Reactor Control Room ventilation air intake.

APPLICABILITY: ALL MODES

ACTION:

With the chlorine detection system inoperable, initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation; restore the chlorine detection system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.6 The chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	≤ 0.10 ppm	≤ 1.00 ppm
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

ECCS SUBSYSTEMS - ~~T_{avg} ≥ 280°F~~

LIMITING CONDITION FOR OPERATION

3.5.2 Two ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and automatically transferring suction to the containment sump on a borated water storage tank low level signal during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

c. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the DHR system from the Reactor Coolant System when the Reactor Coolant System pressure is $\geq$ 280 psig.

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ECCS SUBSYSTEMS - ~~T_{avg} [illegible] 280°F~~

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high pressure injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) and transferring suction to the containment emergency sump.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the HPI pump or the flow path from the borated water storage tank, restore at least one ECCS subsystem to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the decay heat cooler or LPI pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 280°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

EMERGENCY CORE COOLING SYSTEMS

BORATED WATER STORAGE TANK

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LIMITING CONDITION FOR OPERATION

3.5.4 The borated water storage tank (BWST) shall be OPERABLE with:

a. A contained borated water volume of between 402,500 and (   ) gallons,

b. Between 1800 and ~~2000~~ 2200 ppm of boron, and

c. A minimum water temperature of 35°F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The BWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank,

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the water temperature when outside air temperature <35°F.

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LIMITING CONDITION FOR OPERATION

3.5.4 The borated water storage tank (BWST) shall be OPERABLE with:

a. A contained borated water volume of between 402,500 and ( ) gallons,

b. Between 1800 and ~~2000~~ 2200 ppm of boron, and

c. A minimum water temperature of 35°F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The BWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank,

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the water temperature when outside air temperature <35°F.

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CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between +.5" H_2O and -5" H_2O ~~psig~~. from shield bldg pressure

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to within the limits at least once per 12 hours.

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TABLE 3.6-2

CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME (seconds)
30	DH9A	Containment Sump Emergency Recirc Line	No	71
31	DH9B	Containment Sump Emergency Recirc Line	No	71
32	RC1773A	RCS Drain to RC Drain Tank	Yes	10
32	RC1773B	RCS Drain to RC Drain Tank	Yes	10
~~35~~	~~AF599~~	~~Auxiliary Feedwater Line~~	~~Yes~~	~~10~~
~~36~~	~~AF608~~	~~Auxiliary Feedwater Line~~	~~Yes~~	~~10~~
37	FW601	Main Feedwater Line	No	15
38	FW612	Main Feedwater Line	No	10
39	MS100	Main Steam Line	No	10
39	MS107	Main Steam Line	Yes	10
39	MS107A	Main Steam Line	Yes	10
39	ICS11A	Main Steam Line	No	10
39	MS375	Main Steam Line	Yes	10
39	MS100A	Main Steam Line	Yes	10
40	MS101	Main Steam Line	No	10
40	MS106	Main Steam Line	Yes	10
40	MS106A	Main Steam Line	Yes	10
40	ICS11B	Main Steam Line	No	10
40	MS394	Main Steam Line	Yes	10
40	MS101A	Main Steam Line	Yes	10
41	RC232	Pressurizer Quench Tank Circulating Inlet Line	Yes	10
42A	SA2010	Service Air Supply Line	Yes	10
42B	CV5010E	Containment Vessel Air Sample Return	Yes	10
43A	IA2011	Instrument Air Supply Line	No	10
43B	CV5011E	Containment Vessel Air Sample Return	Yes	10
44A	CF1541	Core Flood Tank Fill and N2 Supply Line	Yes	10
44B	NN235	Pressurizer Quench Tank N2 Supply Line	Yes	10

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME (seconds)
74B	CV5010D	Containment Air Sample	Yes	10
74B	CV5011D	Containment Air Sample	Yes	10
74C	DH2735	Pressurizer Auxiliary Spray	Yes	10
74C	DH2736	Pressurizer Auxiliary Spray	Yes	10
B.	CONTAINMENT PURGE AND EXHAUST ISOLATION			
33	CV5005	Containment Vessel Purge Inlet Line	Yes	10
33	CV5006	Containment Vessel Purge Inlet Line	Yes	10
34	CV5007	Containment Vessel Purge Outlet Line	Yes	10
34	CV5008	Containment Vessel Purge Outlet Line	Yes	10
C.	MANUAL			
17	CV343	Containment Vessel Leak Test Inlet Line	No	N/A
23	SF1	Fuel Transfer Tube	No	N/A
24	SF2	Fuel Transfer Tube	No	N/A
*25	CS33	Containment Spray Line	Yes	N/A
*25	CS17	Containment Spray Line	Yes	N/A
*25	SA536	Containment Spray Line	Yes	N/A
*25	SA532	Containment Spray Line	Yes	N/A
*26	CS36	Containment Spray Line	Yes	N/A
*26	CS18	Containment Spray Line	Yes	N/A
*26	SA535	Containment Spray Line	Yes	N/A
29	DH21	Decay Heat Pump Suction Line	No	N/A
29	DH23	Decay Heat Pump Suction Line	No	N/A
*47A	CF2A**	Core Flood Tank Sample Line	Yes	N/A
*47A	CF2B**	Core Flood Tank Sample Line	Yes	N/A
35	AF599	AUX FEED WATER LINE	NO	N/A
36	AF608	" " " "	NO	N/A

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3/4.6.5 SECONDARY CONTAINMENT

EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent emergency ventilation subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one emergency ventilation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each emergency ventilation ~~sub~~system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Initiating from the control room flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least ~~10 hours with the heaters on~~, ~~and~~ 15 minutes

2. Verifying that each ventilation subsystem is aligned to receive electrical power from separate OPERABLE ~~emergency~~ essential busses.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the filter train starts automatically on any containment isolation test signal.

4. Verifying that the filter cooling bypass valves can be manually opened.

5. Verifying that each system produces a negative pressure of ≥ (0.25) inches W.G. in the annulus at a system flow rate of 8000 cfm ± 10%, within (1) minute after a start signal.

15

6. ~~Verifying that the heaters dissipate ≥ ___ ± ___ kw when tested in accordance with ANSI N510-1975.~~

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the filter system at a flow rate of 8000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filter system at a flow rate of 8000 cfm ± 10%.

CONTAINMENT SYSTEMS

SHIELD BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.2 SHIELD BUILDING INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

3.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated at least once per 31 days by verifying that airtight doors and the blowout panels listed in Table 4.61 are closed except when the airtight doors are being used for normal transit entry and exit.

4.6-1

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CONTAINMENT SYSTEMS

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.3 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.~~X~~.3.

6

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the shield building not conforming to the original acceptance standards, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the shield building detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1.

TABLE 4.6-1

ACCESS OPENINGS REQUIRED TO BE CLOSED TO ENSURE SHIELD BUILDING INTEGRITY

I. AIR TIGHT DOORS

DOOR NO.	DESCRIPTION	ELEVATION
100	Access Door from the No. 1 ECCS Pump Room (Room 105) to Pipe Tunnel 101	545'
104A	Access Door from Stair AB-3 to the No. 1 ECCS Pump Room (Room 105)	555'
105	Access Door from Passage 110A to the area above the Decay Heat Coolers	555'
107	Access Door from the No. 2 ECCS Pump Room (Room 115) to the MIscellaneous Waste Monitor Tank and Pump Room (Room 114)	555'
108	Access Door from the No. 2 ECCS Pump Room (Room 115) to the Detergent Waste Drain Tank and Pump Room (Room 125)	555'
201-A	Access Door from Corridor 209 to the No. 1 Mechanical Penetration Room (Room 208)	565'
204	Access Door from Passage 227 to the Makeup Pump Room (Room 225)	565'
205	Access Door from Passage 227 to the No. 2 Mechanical Penetration Room (Room 236)	585'
308	Access Door from Corridor 304 to the No. 4 Mechanical Penetration Room (Room 314)	585'

II. BLOWOUT PANELS

TOTAL NO.	LOCATION	ELEVATION
1	No. 2 Mechanical Penetration Room (Room 236)	565'
6	No. 3 Mechanical Penetration Room (Room 303)	585'
6	No. 4 Mechanical Penetration Room (Room 314)	585'

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided:

a. The inoperable isolation valve is maintained closed.

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

7.5

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3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 Two independent control room emergency ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is $\leq$ (120)°F. delete

b. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes and

2. Verifying that each ventilation system is aligned to receive electrical power from separate OPERABLE essential busses.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

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GROUP HEIGHT - AXIAL POWER SHAPING ROD GROUP

LIMITING CONDITION FOR OPERATION

3.1.3.2 All axial power shaping rods (APSR) shall be OPERABLE, unless fully withdrawn, and shall be positioned within ± 6.5% (indicated position) of their group average height.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one APSR inoperable or misaligned from its group average height by more than ± 6.5% (indicated position), operation may continue provided that within 2 hours:

a. The APSR group is positioned such that the misaligned rod is restored to within limits for the group average height, or

b. It is determined that the imbalance limits of Specification 3.2.1 are satisfied and movement of the APSR group is prevented while the rod remains inoperable or misaligned.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 The position of each APSR rod shall be determined to be within the group average height limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Asymmetric ~~Control~~ Rod ~~Monitor~~ Fault Circuitry is inoperable, then verify the ~~individual rod positions~~ ROD WITH THE INOPERABLE FAULT CIRCUITRY TO BE WITHIN THE LIMIT at least once per 4 hours.

4.1.3.2.2 Unless all APSR are fully withdrawn, each APSR shall be determined to be OPERABLE by moving the individual rod at least 2% at least once every 31 days.

*See Special Test Exceptions 3.10.1 and 3.10.2.

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REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All safety, regulating and axial power shaping control rod absolute position indicator channels and relative position indicator channels shall be OPERABLE and capable of determining the control rod positions within ± 6.5%.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one absolute position indicator channel per control rod group or one relative position indicator channel per control rod group inoperable, within 8 hours reduce THERMAL POWER to ≤ 60% of the THERMAL POWER allowable for the reactor coolant pump combination and reduce the Nuclear Overpower Trip Setpoint to ≤ 70% of the THERMAL POWER allowable for the reactor coolant pump combination.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each absolute and relative position indicator channel shall be determined to be OPERABLE by verifying that the absolute position indicator channels and the relative position indicator channels agree with 6.5% at least once per 12 hours except during time intervals when the Asymmetric Rod ~~Monitor~~ Fault Circuitry is inoperable, then compare the absolute position indicator and relative position indicator channels at least once per 4 hours.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual safety and regulating rod drop time from the fully withdrawn position shall be $\leq$ 1.66 seconds from power interruption at the control rod drive breakers to 3/4 insertion (25% zone light position) with:

a. $T_{avg} \geq 525°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any safety or regulating rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 and 2.

b. With the rod drop times within limits but determined with less than 4 reactor coolant pumps operating, operation may proceed provided that THERMAL POWER is restricted to less than or equal to the THERMAL POWER allowable for the reactor coolant pump combination operating at the time of rod drop time measurement.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of safety and regulating rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once every 18 months.

SAFETY ROD ~~INSERTION LIMIT~~ withdrawal

LIMITING CONDITION FOR OPERATION

3.1.3.5 All safety rods shall be fully withdrawn.

APPLICABILITY: 1* and 2*#.

ACTION:

With a maximum of one safety rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each safety rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any regulating rod during an approach to reactor criticality.

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.1 and 3.10.2.
#With $K_{eff} \geq$

REGULATING ROD ~~INSERTION~~ withdrawal LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be limited in withdrawal physical ~~insertion~~ as shown on Figures 3.1-1 and 3.1-3 with a rod group overlap of 25 ± 5% between sequential withdrawn groups 5 and 6/7.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits, or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figures within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.1 and 3.10.2.
#With $K_{eff} \geq 1.0$.

withdrawal

REGULATING ROD ~~INSERTION~~ LIMITS

SURVEILLANCE REQUIREMENTS

withdrawal

4.1.3.6 The position of each regulating group shall be determined to be within the ~~insertion~~, sequence and overlap limits at least once every 12 hours except when:

a. The regulating rod insertion limit alarm is inoperable, then verify the groups to be within the insertion limits at least once per 4 hours;

b. The control rod drive sequence alarm is inoperable, then verify the groups to be within the sequence and overlap limits at least once per 4 hours.

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Figure 3.1-2

REGULATING Rod Group Withdrawal Limits for 4 Pump Operation up to Control Rod Interchange (250 ± 10 EFPD)

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Figure 3.1-4

Control Rod Core Location and
Group Assignments up to (~~200 ± 10~~) EFPD
200 ± 10

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Figure 3.1-5

Control Rod Core Location and Group
Assignments after (~~250 ± 10~~) EFPD
200±10

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3/4.2 POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1 and 3.2-2.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION:

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a. Restore the AXIAL POWER IMBALANCE to within its limits within 15 minutes, or

b. Be in at least HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

alarm

4.2.1 The AXIAL POWER IMBALANCE shall be determined to be within limits in each core quadrant at least once every 12 hours when above 40% of RATED THERMAL POWER except when an AXIAL POWER IMBALANCE ~~monitor~~ is inoperable, then calculate the AXIAL POWER IMBALANCE ~~in each core quadrant with an inoperable monitor~~ at least once per hour.

* See Special Test Exception 3.10.1

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AXIAL POWER IMBALANCE ENVELOPE
FOR OPERATION UP TO (400 ± 10) EFPD

Figure 3.2.1

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AXIAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER (400 ± 10) EFPD

Figure 3.2-2

SURVEILLANCE REQUIREMENTS (Continued)

a. Prior to initial operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured F_Q of 4.2.2.1 above, shall be increased by ~~3%~~ 1.4 to account for manufacturing tolerances and further increased by ~~5%~~ to account for measurement uncertainty. 7.5

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

a. Prior to operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% 5% for measurement uncertainty.

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT shall not exceed 4%. 4.92%

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER.*

ACTION:

a. With QUADRANT POWER TILT determined to exceed 4% 4.92% but $\leq$ 9% 11.07%

1. Within 2 hours:

a) Either reduce the QUADRANT POWER TILT to within its limit, or

b) Reduce THERMAL POWER so as not to exceed THERMAL POWER, including power level cutoff, allowable for the reactor coolant pump combination less at least 2% for each 1% of ~~indicated~~ QUADRANT POWER TILT in excess of 4% 4.92 and within 4 hours, reduce the High Flux Trip Setpoint and the Flux - Δ Flux - Flow Trip Setpoint at least 2% for each 1% of indicated QUADRANT POWER TILT in excess of 4% 4.92

2. Verify that the QUADRANT POWER TILT is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the High Flux Trip Setpoint to $\leq$ 55% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above [illegible] of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.1.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT shall be determined to be within the limits at least once every 7 days during operation above 15% of RATED THERMAL POWER except when the QUADRANT POWER TILT ~~monitor~~ is inoperable, then the QUADRANT POWER TILT shall be calculated at least once per 12 hours.

alarm

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure $\leq$ 1820 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER,

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of $\leq$ 1820 psig is imposed, and

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(3) The Shutdown Bypass is removed when RCS pressure > 1900 psig.

ACTION STATEMENTS

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

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1,

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(7)	M	1, 2
3. RC High Temperature	S	R	M	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(7,8)	M	1, 2
5. RC Low Pressure	S	R	M	1, 2
6. RC High Pressure	S	R	M	1, 2
7. RC Pressure-Temperature	S	R	M	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	R	M	1, 2
9. Containment High Pressure	S	R	M	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(5)	1, 2 and*
11. Source Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	2, 3, 4 and 5
~~12. Control Rod Drive Trip Breakers~~	~~N.A.~~	~~N.A.~~	~~M and S/U(1)~~	~~1, 2 and*~~
13. Reactor Trip Module Logic	N.A.	N.A.	M	1, 2 and*
14. Shutdown Bypass High Pressure	S	R	M	3, 4 and 5**

TABLE 4.3-1 (Continued)

NOTATION

* - With control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL P[illegible]R. The CHANNEL FUNCTIONAL TEST shall be excluded from this CALIBRATION

(3) - Compare incore to excore measured AXIAL POWER IMBALANCE above ~~15%~~ 30% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ (~~2~~) 3.5 percent. The Channel Functional Test shall be excluded from this Calibration

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - Verify at least one decade overlap if not verified in previous 7 days.

(6) - Each train tested every other month.

(7) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(8) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However each sensor shall be calibrated at least once per 18 months.

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TABLE 3.3-3

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF LOGIC CHANNELS	LOGIC CHANNELS TO TRIP	MINIMUM LOGIC CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. High Pressure Injection					
1) Containment Pressure - High	4	2	3	1, 2, 3	~~8~~ 9
2) RCS Pressure - Low	4	2	3	1, 2, 3	~~8~~ 9
b. Low Pressure Injection					
1) Containment Pressure - High	4	2	3	1, 2, 3	~~8~~ 9
2) RCS Pressure--Low-Low**	4	2	3	1, 2, 3	~~8~~ 9
2. CONTAINMENT SPRAY					
a. Containment Pressure--High-High	4	2	3	1, 2, 3	~~8~~ 9
3. CONTAINMENT ISOLATION					
a. Containment Pressure - High	4	2	3	1, 2, 3	~~8~~ 9
b. Containment Pressure--High-High	4	2	3	1, 2, 3	~~8~~ 9
c. Containment Radioactivity - High	4	2	3	ALL MODES	~~8 and 9~~ 9+10
d. RCS Pressure - Low*	4	2	3	1, 2, 3	~~8~~ 9

TABLE 3.3-3 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF LOGIC CHANNELS	LOGIC CHANNELS TO TRIP	MINIMUM LOGIC CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. CONTAINMENT ISOLATION (continued)					
e. RCS Pressure--Low-Low**	4	2	3	1, 2, 3	~~8~~ 9
4. CONTAINMENT COOLING					
a. Containment Pressure - High	4	2	3	1, 2, 3	~~8~~ 9
b. Containment Pressure--High-High	4	2	3	1, 2, 3	~~8~~ 9
c. RCS Pressure-Low	4	2	3	1, 2, 3	~~8~~ 9
5. MAIN STEAM AND FEEDWATER ISOLATION					
a. Containment Pressure--High-High	4	2	3	1, 2, 3	~~8~~ 9
6. CONTAINMENT SUMP SUCTION					
a. Borated Water Storage Tank - Low	4	2	3	1, 2, 3	~~8~~ 9

TABLE 3.3-3 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. SFAS MANUAL ACTUATION - CHANNEL A	1	1	1	1, 2, 3, 4	~~10~~ 11
8. SFAS MANUAL ACTUATION - CHANNEL B	1	1	1	1, 2, 3, 4	~~10~~ 11
9. CONTAINMENT SPRAY MANUAL ACTUATION CHANNEL 1	1	1	1,	1, 2, 3	~~10~~ 11
10. CONTAINMENT SPRAY MANUAL ACTUATION CHANNEL 2	1	1	1	1, 2, 3	~~10~~ 11
11. COINCIDENCE LOGIC CHANNELS	34	2***	2***	All Modes	~~11~~ 12
12. SEQUENCE LOGIC CHANNELS	4	2	3	1, 2, 3	~~8~~ 9

TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1800 psig. Bypass shall be automatically removed when RCS pressure exceeds 1800 psig.

** Trip function may be bypassed in this MODE with RCS pressure below 600 psig. Bypass shall be automatically removed when RCS pressure exceeds 600 psig.

ACTION STATEMENTS

ACTION ~~8~~ 9 - With the number of OPERABLE channels one less than the Total Number of Channels and with RC system pressure.

a. < 1800 psig, place the inoperable channel in the bypassed condition within one hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing the RCS pressure above 1800 psig; otherwise be in at least HOT STANDBY within the following 6 hours.

b. $\geq$ 1800 psig, place the inoperable channel in the bypassed condition within one hour; operation may continue provided that the Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.2.

ACTION ~~9~~ 10 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust isolation valves are maintained closed.

ACTION ~~10~~ 11 - With less than the Minimum Channels OPERABLE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION ~~11~~ 12 - With any component in the coincidence logic channels inoperable, trip that component within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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Figure 2.1-2

Reactor Core Safety Limit

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. High Flux	$\leq$ 105.44% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating
See previously accepted deviation from STS. DB system does not work this way!	$\leq$ 80.7% of RATED THERMAL POWER with three pumps operating	$\leq$ (80.7%) of RATED THERMAL POWER with three pumps operating
	$\leq$ 53.0% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ (53.0%) of RATED THERMAL POWER with one pump operating in each loop
3. RC High Temperature	$\leq$ 618.95°F	$\leq$ 619°F
4. Flux - Δ Flux-Flow(1)	Trip Setpoint adjusted to not exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2.
5. RC Low Pressure(1)	$\geq$ 1985.4 psig	$\geq$ 1985 psig
6. RC Low High Pressure(1)	$\leq$ 2356 psig 2354.6	$\leq$ 2355 psig
7. RC Pressure-Temperature(1)	$\geq$ (13.85 T_{out} °F - 6494) psig	$\geq$ (13.85 T_{out} °F - 6498) psig

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TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. High Flux[(1)]	$\leq$ 105.44% of RATED THERMAL POWER	$\leq$ 105.5% of RATED THERMAL POWER
3. RC High Temperature	$\leq$ 618.95°F	$\leq$ 619°F
4. Flux- Δ Flux - Flow[(2)]	Trip Setpoint adjusted to not exceed the limit lines of Figure 2.2-1	Allowable Values not to exceed the limit lines of Figure 2.2-2.

---

(1) When the High Flux trip setpoint is required to be reduced by an ACTION statement to some percentage of the THERMAL POWER allowable for the reactor coolant pump combination, the THERMAL POWER allowable:

a. For 3 pump operation, is 80.7% of RATED THERMAL POWER.
b. For operation with one pump in each loop is 53% of RATED THERMAL POWER.

(2) Trip may be manually bypassed when RCS pressure $\leq$ 1820 psig by actuating Shutdown Bypass provided that:

a. The ~~Nuclear Overpower~~ High FLUX trip set point is $\leq$ 5% of RATED THERMAL POWER.
b. The Shutdown Bypass High Pressure trip is automatically imposed with a set point $\leq$ 1820 psig, and
c. The Shutdown Bypass is manually removed when RCS pressure > ~~1900~~ 1985.4.

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Figure 2.2-1

~~Trip Setpoint for Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE~~

Flux – Δ Flux FLOW TRIP SETPOINTS

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Figure 2.2-2

~~Allowable Value for Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE~~

Flux - ΔFlux - Flow Trip Allowable Values

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2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the B&W-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.32. This value corresponds to a 95 percent probability at a 99 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.32 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 131.3×10^6 lbs/hr, which is the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F_q \quad \cancel{F_Q^N} = 2.56; \quad F_{\Delta H}^N = 1.71; \quad F_Z^N = 1.50$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

BASES

The reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about ( ) psi less core outlet pressure, providi a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.32 DNBR limit produced by a nuclear power peaking factor of ~~F_q^N~~ = 2.~~59~~ or the combination of the radial peak,
$F_q = 2.56$ axial peak and position of the axial peak that yields no less than a 1.32 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.4 kw/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.32 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to +22%, whichever condition is more restrictive.

Using a local quality limit of +22% at the point of minimum DNBR as a basis for curve 3 of BASES Figure 2.1 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the B&W-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of +22% is justified on the basis of experimental data.

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.32 or a local quality at the point of minimum DNBR less than +22% for that particular reactor coolant pump situation. The 1.32 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Trip Setpoint specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip setpoint less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Shutdown Bypass provides for bypassing certain functions of the Reactor Protection System in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the Shutdown Bypass High Pressure trip is to prevent normal operation with Shutdown Bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The High Flux Trip Setpoint of $\leq 5.0\%$ prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic Reactor Protection System instrumentation channels and provides manual reactor trip capability.

High Flux

A High Flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

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RC High Temperature

The RC High Temperature trip $\leq$ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux - Δ Flux-Flow

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the High Flux/Number of Reactor Coolant Pumps On Trips.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.2-1 are as follows:

at the allowable value of

1. Trip would occur when four reactor coolant pumps are operating if power is 108.0% and reactor flow rate is 100%, or flow rate is 92.6% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.7% and reactor flow rate is 74.7%, or flow rate is 69.4% and power is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.9% and reactor flow rate is 49.0% or flow rate is 45.4% and the power level is 49.0%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by ~~1.08%~~ an allowable value of 1.08% for a 1% flow reduction.

RC Pressure - Low, High and Pressure Temperature

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC High Pressure setpoint is reached before the High Flux Trip Setpoint. The trip setpoint for RC High Pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC High Pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2435 psig. The RC High Pressure trip also backs up the High Flux trip.

The RC Low Pressure, 1935 psig, and RC Pressure-Temperature (13.85 Tout°F-6498) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.32 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RC Pressure-Temperature Trip Setpoint of ((13.85) Tout°F-(6498)) psig. delete

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the Flux - Δ Flux-Flow trip the High Flux/Number of Reactor Coolant Pumps On trip prevents the minimum core DNBR from decreasing below 1.32 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

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BASES Figure 2.1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN < 1% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 18 gpm of 7875 ppm boron or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours, by verifying that regulating rod groups withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2## within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of e. below, with the regulating rod groups at the ~~maximum insertion~~ limit of Specification 3.1.3.6. minimum withdrawal

#With $K_{eff} \geq 1.0$.

##With $K_{eff} < 1.0$.

*See Special Test Exception 3.10.4.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest loop temperature (T_{avg}) shall be $\geq$ 525°F.

APPLICABILITY: MODES 1 and 2*. ※

ACTION:

With a Reactor Coolant System loop temperature (T_{avg}) < 525°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The RCS temperature (T_{avg}) shall be determined to be $\geq$ 525°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than ~~525~~°F.
530°

*With $K_{eff} \geq 1.0$.

※ See special Test Exception 3.10.2.

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 At least one of the following boron injection flow paths shall be OPERABLE.

a. A flow path from the ~~concentrated~~ boric acid addition ~~storage~~ system via a boric acid pump and a makeup or decay heat removal (DHR) pump to the Reactor Coolant System, if only the boric acid addition ~~storage~~ system in Specification 3.1.2.8a is OPERABLE, or

b. A flow path from the borated water storage tank via a makeup or DHR pump to the Reactor Coolant System if only the borated water storage tank in Specification 3.1.2.8b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pipe temperature of the heat traced portion of the flow path is $\geq$ 105°F when a flow path from the concentrated boric acid storage system is used, and

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the ~~concentrated~~ boric acid addition ~~storage~~ system via a boric acid pump and makeup or decay heat removal (DHR) pump to the Reactor Coolant System, and

b. A flow path from the borated water storage tank via makeup or DHR pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the concentrated boric acid storage system inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the borated water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVIELLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pipe temperature of the heat traced portion of the flow path from the concentrated boric acid storage system is $\geq$ 105°F.

DECAY HEAT REMOVAL PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one decay heat removal (DHR) pump in the boron injection flow path required by Specification 3.1.2.1 or 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE ~~emergency~~ essential bus.

APPLICABILITY: MODES 4,* 5* and 6.

ACTION:

With no DHR pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one DHR pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*RCS Pressure < 300 psig.

BORIC ACID PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid essential pump shall be OPERABLE and capable of being powered from an OPERABLE ~~emergency~~ bus if only the flow path through the boric acid pump in Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least one boric acid pump essential in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE ~~emergency~~ bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least one boric acid pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.7 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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LIMITING CONDITION FOR OPERATION

3.1.2.9 Each of the following borated water sources shall be OPERABLE:

a. The concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume in accordance with Figure 3.1-1,

2. Between 7875 and 13,125 ppm of boron, and

3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A contained borated water volume of between 402,500 and ________ gallons,

2. Between 1800 and ___ ppm of boron, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

addition

a. With the ~~concentrated~~ boric acid ~~storage~~ system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F within the next 6 hours; restore the concentrated boric accid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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Regulatory Docket File

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified as a reportable occurrence in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

 1. Capable of being closed by the SFAS ~~an OPERABLE containment automatic isolation system~~, or

 b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.1.

b. All equipment hatches are closed and sealed,

c. Each airlock is OPERABLE pursuant to Specification 3.6.1.3, and

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

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

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be 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 instrument channels measuring the same parameter.

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied ~~to~~ FROM the reactor coolant pump seals.

QUADRANT POWER TILT

1.18 QUADRANT POWER TILT shall be defined by the following equation ~~the maximum difference between the power generated in any core quadrant (upper or lower core half) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core~~ and is expressed in percent.

~~Quadrant Power Tilt~~

$$= 100 \cancel{\max} \left(\frac{\text{Power in any core quadrant } \cancel{\text{(upper or lower)}}}{\text{Average power of all quadrants } \cancel{\text{(upper or lower)}}} - 1 \right)$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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."

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\overline{E}$-AVERAGE DISINTEGRATION ENERGY shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies

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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.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

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

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

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

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL POWER IMBALANCE

1.23 AXIAL POWER IMBALANCE shall be the THERMAL POWER in the top half of the core expressed as a percentage of RATED THERMAL POWER minus the THERMAL POWER in the bottom half of the core expressed as a percentage of RATED THERMAL POWER ~~for each quadrant~~.

SHIELD BUILDING INTEGRITY

1.24 SHIELD BUILDING INTEGRITY shall exist when:

a. The airtight doors and the blowout panels listed in Table 4.6-1 are closed except the airtight doors may be used for normal transit entry and exit.

b. The emergency ventilation system is OPERABLE.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.25 The REACTOR PROTECTION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers.

THIS P[illegible] RECEIPT OF
INFOR[illegible] FROM THE APPLICANT
NOT TRUE

Figure 2.1-1

Reactor Core Safety Limit