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**Washington Public Power Supply System**
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

March 17, 1982
G02-82-327



Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
PARTIAL RESUBMITTAL OF DRAFT
TECHNICAL SPECIFICATIONS

Reference: Letter, G02-82-130, G.D. Bouchey (SS)
to A. Schwencer (NRC), Same Subject,
dated February 1, 1982

Due to duplication errors, we are resubmitting two (2) sections of the draft Technical Specifications for WNP-2, transmitted per referenced letter. Enclosed are ten (10) copies for your review.

Please excuse any inconvenience our error may have caused your office.

Very truly yours,

G D Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

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8203240198

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

Technical Specifications

Appendix "A"
to
License No. ____

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ADMINISTRATIVE CONTROLS

[illegible]

## 1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the 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. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 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.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

It is difficult to measure RPT Breaker position, therefore we must measure coil energization time. The clarifing information on turbine throttle valves and turbine governor valves movement definitions are required to ensure proper response time measurement.

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## DEFINITIONS

### CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position. Normal movement of control rod with the CRD hydraulic system and normal TIP/SRM/IRM movement are not considered core alterations.

### CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

### DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries 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."

### ~~Ē-AVERAGE DISINTEGRATION ENERGY~~

delete

~~1.10 Ē shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.~~

### EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.10 ~~1.11~~ The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable.

### END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.11 ~~1.12~~ The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump ~~circuit~~ breaker trip coil energization from initial movement of the associated:

a. Turbine throttle ~~stop~~ valves, ~~and~~ channel sensor contact opening and,

b. Turbine governor ~~control~~ valves initiation of valve fast closure.



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IDENTIFIED and UNIDENTIFIED Leakage changes

Unidentified leakage within the primary containment enters the floor drain sump only. It is possible that some "IDENTIFIED" leakage such as an RBCCW cooling flow to the recirc. pump air coolers leak, could ALSO enter the floor drain system. If the [illegible] of this leakage term is determined and satisfies the IDENTIFIED definition this leakage should <u>NOT</u> be included in the UNIDENTIFIED leakage measured through the FD system i.e.

unidentified leakage, FDS flow − "identified" $\leq$ 5 gpm $\Rightarrow$

FDS flow $\leq$ 5 gpm + "identified"

and total leakage, EDS flow + FDS flow $\leq$ 25 gpm

$\Rightarrow$ EDS flow $\leq$ 25 gpm − FDS flow

## DEFINITIONS

~~FRACTION OF LIMITING POWER DENSITY~~

1.12 ~~1.~~ ~~The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the limiting LHGR for that bundle type.~~

~~FRACTION OF RATED THERMAL POWER~~

1.13 ~~1.~~ ~~The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.~~

FREQUENCY NOTATION

1.14 ~~1.13~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 ~~1.14~~ IDENTIFIED LEAKAGE shall be:

a. Leakage into the Drywell Equipment Drain collection sytem~~s~~, such as pump seal or valve packing leaks, that is captured and conducted ~~to a sump or collecting tank, or~~

b. Leakage into the ~~containment atmosphere~~ Drywell Floor Drain Collection system from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.16 ~~1.15~~ The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable.

LIMITING CONTROL ROD PATTERN

1.17 ~~1.16~~ A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.18 ~~1.17~~ LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod, Expressed in units of KW/ft. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.19 ~~1.18~~ A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, ie., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY.

## DEFINITIONS

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.20 ~~1.~~ The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be highest value of the FLPD which exists in the core.

MAXIMUM TOTAL PEAKING FACTOR

1.21 ~~1.19~~ The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MINIMUM CRITICAL POWER RATIO

1.22 ~~1.20~~ The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OPERABLE - OPERABILITY

1.23 ~~1.21~~ A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.24 ~~1.22~~ An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2..

PHYSICS TESTS

1.25 ~~1.23~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.26 ~~1.24~~ PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall. Leakage directed to the Drywell Equipment Drain Sump is specifically excluded.

[illegible]

PRIMARY CONTAINMENT INTEGRITY

(1.27) ~~1.25~~ PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification, 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

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

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

RATED THERMAL POWER

(1.28) ~~1.26~~ RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~(2436)~~ MWT. (3323)

REACTOR PROTECTION SYSTEM RESPONSE TIME

(1.29) ~~1.27~~ REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-engergization of the scram pilot valve solenoids.

REPORTABLE OCCURRENCE

(1.30) ~~1.28~~ A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

ROD DENSITY

(1.31) ~~1.29~~ ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SOURCE CHECK

1.34 a SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source

## DEFINITIONS

### SECONDARY CONTAINMENT INTEGRITY

1.32 ~~1.30~~ SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification ~~3.6.5.2.~~

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

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

f. The capability exists to establish pressure within the secondary containment ~~is~~ at less than or equal to 0.25 inches of vacuum water gauge.

### SHUTDOWN MARGIN

1.33 ~~1.31~~ SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

← add →

### STAGGERED TEST BASIS

1.35 ~~1.32~~ A STAGGERED TEST BASIS shall consist of:

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

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

### THERMAL POWER

1.36 ~~1.33~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

## DEFINITIONS

TOTAL PEAKING FACTOR

1.37 ~~1.34~~ The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the core average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.38 ~~1.35~~ The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ~~its~~ isolation actuation setpoint at the channel sensor until the turbine bypass valves travel to their required positions.

UNIDENTIFIED LEAKAGE

1.39 ~~1.36~~ UNIDENTIFIED LEAKAGE shall be all leakage into the DRYWELL Floor Drain System, which is not IDENTIFIED LEAKAGE.

VENTING

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



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TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

| NOTATION | FREQUENCY |
|---|---|
| S | At least once per 12 hours. |
| D | At least once per 24 hours. |
| W | At least once per 7 days. |
| M | At least once per 31 days. |
| Q | At least once per 92 days. |
| SA | At least once per 184 days. |
| A | At least once per 366 days. |
| R | At least once per 18 months (550 days). |
| S/U | Prior to each reactor startup. |
| N.A. | Not applicable. |

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TABLE 1.2

OPERATIONAL CONDITIONS

| CONDITION | MODE SWITCH POSITION | AVERAGE REACTOR COOLANT TEMPERATURE |
|---|---|---|
| 1. POWER OPERATION | Run | Any temperature |
| 2. STARTUP | Startup/Hot Standby | Any temperature |
| 3. HOT SHUTDOWN | Shutdown# | > 200°F |
| 4. COLD SHUTDOWN | Shutdown#,## | $\leq$ 200°F |
| 5. REFUELING* | Shutdown or Refuel**,# | $\leq$ ~~140°F~~ 212°F |

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified ~~member of the unit technical staff.~~ person

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exception 3.10.3

[illegible] 01 [illegible]

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06 and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ~~low pressure~~ ECCS to restore the water level. ~~after depressurizing the reactor vessel, if required.~~

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

The basis for the instrument setpoints are discussed in a GE document written for WNP-2 by GE entitled "Instrument Setpoints and Bases for Standard Specifications,

Fuel safety limits and the bases for section 2.0 rely on two documents:

1. NEDO-10958-A (GETAB)
2. NEDO-20944-P (BWR/5 Fuel Design)

Specific reference to supporting documentation is provided on back table.

Four "laters" appear in Table 2.2.1-1;

Two (2) Scram Discharge Volume High - Determined in start-up
Two (2) TCV Fast closure Trip Oil Pressure Low - Data unavailable

These setpoints can not be determined at present.

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## TABLE 2.2.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Intermediate Range Monitor, Neutron Flux-High | ≤ 120 divisions of full scale | ≤ 122 divisions of full scale |
| 2. Average Power Range Monitor: | | |
| a. Neutron Flux-~~Upscale (Setdown)~~ HIGH | ≤ 15% of RATED THERMAL POWER | ≤ 20% of RATED THERMAL POWER |
| b. Flow Biased Simulated Thermal Power-~~Upscale~~ | | |
| 1) Flow Biased | ≤ 0.66 W+51%, with a maximum of | ≤ 0.66 W+54%, with a maximum of |
| 2) High Flow Clamped | ≤ 113.5% of RATED THERMAL POWER | ≤ 115.5% of RATED THERMAL POWER |
| c. Fixed Neutron Flux-~~Upscale, (118)%~~ HIGH | ≤ 118% of RATED THERMAL POWER | ≤ 120% of RATED THERMAL POWER |
| 3. Reactor Vessel Steam Dome Pressure - High | ≤ (~~1045~~) 1043 psig | ≤ (~~1065~~) 1063 psig |
| 4. Reactor Vessel Water Level - Low, Level 3 | ≥ 12.5 inches above instrument zero* | ≥ 11.0 inches above instrument zero* |
| 5. Main Steam Line Isolation Valve - Closure | ≤ 6% closed | ≤ 7% closed |
| 6. Main Steam Line Radiation - High | ≤ 3.0 x full power background | ≤ 3.6 x full power background |
| 7. Primary Containment Pressure - High | ≤ 1.69 psig | ≤ 1.89 psig |
| 8. Scram Discharge Volume Water Level - High | ≤ (~~50~~) Later gallons | ≤ ( Later ) gallons |
| 9. Turbine Stop Valve - Closure | ≤ 5% closed | ≤ 7% closed |
| 10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low | ≥ (~~500~~) later psig | ≥ ( Later ) psig |
| 11. Reactor Mode Switch Shutdown Position | NA | NA |
| 12. Manual Scram | NA | NA |

*See Bases Figure B 3/4 3-1.



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[illegible]

## 2.1 SAFETY LIMITS

### BASES

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.06. MCPR greater than 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Integrity Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

### 2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of $28 \times 10^3$ lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than $28 \times 10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

## 2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no (mechanistic) fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), GEXL correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

| | |
|---|---|
| Pressure: | 800 to 1400 psia |
| Mass Flow: | $0.1 \times 10^6$ to $1.25 \times 10^6$ lb/hr-ft$^2$ |
| Inlet Subcooling: | 0 to 100 Btu/lb |
| Local Peaking: | *1.61* at a corner rod to<br>*1.47* at an interior rod |

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

[illegible]

THERMAL POWER, High Pressure and High Flow (Continued)

| Axial Peaking: | Shape | Max/Avg. |
|---|---|---|
| | Uniform | 1.0 |
| | Outlet Peaked | 1.60 |
| | Inlet Peaked | 1.60 |
| | Double Peak | 1.46 and 1.38 |
| | Cosine | 1.39 |
| Rod Array | 64 Rods in an 8 x 8 array | |

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1, the nominal values of the core parameters listed in Bases Table B2.1.2-2, and the relative assembly power distribution shown in Bases Table B2.1.2-3. Bases Table B2.1.2-4 shows the R-factor distributions that are input to the statistical model which is used to establish the Safety Limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Admendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

## Bases Table B2.1.2-1

## UNCERTAINTIES USED IN THE DETERMINATION OF THE FUEL CLADDING SAFETY LIMIT*

| Quantity | Standard Deviation (% of Point) |
|---|---|
| Feedwater Flow | 1.76 |
| Feedwater Temperature | 0.76 |
| Reactor Pressure | 0.5 |
| Core Inlet Temperature | 0.2 |
| Core Total Flow | 2.5 |
| Channel Flow Area | 3.0 |
| Friction Factor Multiplier | 10.0 |
| Channel Friction Factor Multiplier | 5.0 |
| TIP Readings | 6.3 |
| R Factor | 1.5 |
| Critical Power | 3.6 |

* The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core.

[illegible]

## Bases Table B2.1.2-2

## NOMINAL VALUES OF PARAMETERS USED IN THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

| | |
|---|---|
| THERMAL POWER | 3323 MW |
| Core Flow | 108.5 Mlb/hr |
| Dome Pressure | 1010.4 psig |
| Channel Flow Area | 0.1089 ft$^2$ |
| R-Factor | High enrichment - 1.043<br>Medium enrichment - 1.039<br>Low enrichment - 1.030 |

[illegible]

## Bases Table B2.1.2-3

## RELATIVE BUNDLE POWER DISTRIBUTION

## USED IN THE GETAB STATISTICAL ANALYSIS

| Range of Relative Bundle Power | Percent of Fuel Bundles Within Power Interval |
|---|---|
| 1.525 to 1.575 | 2.1 |
| 1.475 to 1.525 | 8.9 |
| 1.425 to 1.475 | 9.9 |
| 1.375 to 1.425 | 3.1 |
| 1.325 to 1.375 | 5.2 |
| 1.275 to 1.325 | 2.1 |
| 1.225 to 1.275 | 5.2 |
| 1.175 to 1.225 | 2.1 |
| 1.125 to 1.175 | 6.3 |
| 1.075 to 1.125 | 5.8 |
| 1.025 to 1.075 | 1.0 |
| ≤ 1.025 | 48.3 |
| | 100.0 |

Bases Table B2.1.2-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

8x8 Rod Array

| R-Factor High Enrichment | Medium Enrichment | Low Enrichment | Rod Sequence No. |
|---|---|---|---|
| 1.043 | 1.039 | 1.030 | 1 |
| 1.043 | 1.039 | 1.030 | 2 |
| 1.042 | 1.028 | 1.030 | 3 |
| 1.042 | 1.028 | 1.030 | 4 |
| 1.038 | 1.027 | 1.028 | 5 |
| 1.038 | 1.027 | 1.028 | 6 |
| 1.026 | 1.026 | 1.028 | 7 |
| ≤ 1.024 | ≤ 1.026 | ≤ 1.028 | 8 thru 64 |

→ WNP-2 FSAR Tables 5.2-3, 5.2-5

## SAFETY LIMITS

## BASES

### 2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code 71 Edition, including Addenda through Summer 1971, which permits a maximum pressure transient of (110)%, (~~1375) psig at the lowest elevation of the reactor coolant system~~. The reactor coolant system is designed to the ~~(USAS Piping Code, Section B31.1) (and the)~~ ASME Boiler and Pressure Vessel Code, 71 Edition, including Addenda through Winter 1971 for the reactor recirculation piping, which permits a maximum pressure transient of (~~120~~) 125 %, (~~1380~~) 1565 psig, of design pressure, (~~1150~~) 1250 psig for suction piping and (~~1250~~) 1550 psig for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the (applicable codes).

### 2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

## BASES

### 2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 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 during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have (15.4) been analyzed. The results of these analyses are in Section ~~(15.1.12)~~ of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all the possible sources of reactivity input, uniform control rod

## LIMITING SAFETY SYSTEM SETTINGS

## BASES

### REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

#### Average Power Range Monitor (Continued)

withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux ~~Upscale (118)%~~ setpoint; High i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-~~Upscale~~ High setpoint, ~~a time constant of (6) seconds is introduced in~~to the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when the design TOTAL PEAKING FACTOR is exceeded ~~(MFLPD is greater than or equal to FRTP)~~.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

[illegible] 10 83

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint ~~(has been used in transient analyses dealing with coolant inventory decrease. The results reported in Section 15 show that scram and isolation of all process lines, except main steam, at this level adequately protects the fuel and the pressure barrier, because MCHFR is greater than 1.0 in all cases, and system pressure does not reach the safety valve settings. The scram setting)~~ was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature, and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. ~~(No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.)~~

7. Primary Containment Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trip.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water ~~at pressures below 65 psig~~, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods ~~at pressures below 65 psig~~ when they are tripped.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. (5) With a trip setting of (5)% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient (assuming the turbine bypass valves operate).

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection ~~coincident with~~ failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a faster closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section (~~15.1.0~~) 15.2.2 of the Final Safety Analysis Report.

11. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

[illegible]

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

## 3/4.0 APPLICABILITY

### LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

or the requirements of specification 3.03

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

4.0.2.c → BASIS: Letter; Schwencer to Bolger 11/30/74 with respect to "NRC Staff guidance for complying with certain provisions of 10 CFR 55".

---

4.0.3 "Performance of a surveillance requirement within the specified interval shall constitute compliance with Operability Requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment. Failure to perform a surveillance requirement within the specified time interval shall constitute a failure to satisfy the operability requirements for a Limiting Condition for Operation."

---

replace with this

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c. The surveillance interval of (a) and (b) above may be extended whenever valve cycling could place the plant in an unsafe condition.

APPLICABILITY

## SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirements.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

c.

add 4.03

~~4.0.3~~ Failure to perform a Surveillance Requirement within the ~~specified time~~ interval ~~shall constitute~~ a failure to meet the ~~OPERABILITY~~ requirements for a Limiting Condition ~~for Operation. Exceptions~~ to these requirements are stated in the ~~individual Specificatons.~~ ~~Surveillance~~ requirements do not have to be per~~formed~~ on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.1.1.b → Recommend deleting para 4.1.1.b as it's entirety as it is unnecessarily restrictive. Compliance with the Reactivity Anamoly specification guarantees compliance with shutdown Margin requirements given 4.1.1.a is adequately demonstrated. If NRC rejects this approach, then para 4.1.1.b should be phrased as modified.

4.1.1.c → The time for action in 4.1.1.c should be increased to <u>four</u> hours (at least) as the shutdown margin analysis will take longer than 1 hour to complete.

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## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 SHUTDOWN MARGIN

### LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. ~~(~~0.38~~)~~% delta k/k with the highest worth rod analytically determined, or

b. ~~(~~0.28~~)~~% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS* and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined by measurement or analysis to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

see note — b. ~~By measurement~~, Within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, ~~is equal to~~ most closely approaches the specified limit.

c. Within ~~one~~ four hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

*Except movement of IRMs, SRMs or special movable detectors.

## 3/4.1.2 REACTIVITY ANOMALIES

### LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

### SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

## REACTIVITY CONTROL SYSTEMS

### 3/4.1.3 CONTROL RODS

### CONTROL ROD OPERABILITY

### LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

1. Within one hour:

   a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions, and

   ~~b)~~ Disarm** the associated directional control valves either:
   1) Electrically, or
   2) Hydraulically by closing the drive water and exhaust water isolation valves.

~~2.~~ Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods inoperable for causes other than addressed in ACTION a, above:

1. If the inoperable control rod(s) is withdrawn:

   a) Immediately ~~verify~~:

   verify

   1) That the inoperable withdrawn control rod(s) is separated from all other inoperable control rods by at least two control cells in all directions, and

   demonstrate

   2) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*.

   b) Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves either:

   1) Electrically, or

   2) Hydraulically by closing the drive water and exhaust water isolation valves.

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.



** The associated directional control valves may be re-armed periodically under administrative control to permit maintenance testing of the control rod.

Specification 3.1.3.1 allows up to 8 inoperable control rods which meet the separation criteria and are either demonstrated insertable or fully inserted and disarmed. Fully inserted inoperable control rod drives enhance the EOC scram reactivity since the axial power shape is peaked more to the bottom of the core. Thus current transient analyses bound the case with 8 or fewer fully inserted inoperable control rod drives.

The effect of inoperable rods on the incremental control rod worth was evaluated during the development of the Banked Position withdrawal Sequence. ("Banked Position Withdrawal Sequence", C. S. Paone, December 1977, NEDO-21231). Assuming a worst case distribution of 8 fully inserted, inoperable control rod drives, the highest calculated incremental control rod worth was 1.2% k. This corresponds to a peak fuel enthalpy of 232 cal/gm. which is below the 280 cal/gm. design limit for control rod drop accident.

As long as the Banked Position Withdrawal Sequence is adhered to, shutdown margin requirements are satisified and either the insertability of the inoperable control rod drive is demonstrated or the drive is fully inserted and valved out of service, it is safe to startup with inoperable control rod drives.

Therefore, except for certain cases that indicate problems beyond simple control rod inoperability, exemption from the provisions of Specification 3.0.4 is justified. Revised specifications implementing proposed changes on this topic are attached.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted:

   a) Within one hour disarm the associated directional control valves either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

   b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

See note

3. The provisions of specification 3.0.4 are not applicable.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE at least once per 31 days by:

a. Verifying each valve to be open,* and

b. Cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.



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## SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of at least ~~(~~50~~)~~% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

1. Close within ~~(30)~~ 30 seconds after receipt of a signal for control rods to scram, and

2. Open after the scram signal is reset or the scram discharge volume trip is bypassed.

b. Proper float response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation after each scram from a pressurized condition.



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See justification in specification 3.1.3.1.b.

## CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

### LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 6, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.
The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant ~~pressure~~ pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days.

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

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

*Except movement of SRM, IRM, or special movable detectors or normal control rod movement.



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. INSERTION Times taken from GE Document 22A1342 "Control Rod Drive system spec."

## CONTROL ROD AVERAGE SCRAM INSERTION TIMES

### LIMITING CONDITION FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

| Position Inserted From Fully Withdrawn | Average Scram Insertion Time (Seconds) |
|---|---|
| ~~(46)~~ - 45. | ~~(0.375)~~ 0.430 |
| ~~(38)~~ 39. | ~~(1.096)~~ 0.868 |
| ~~(26)~~ 25. | ~~(2.000)~~ 1.936 |
| ~~(6)~~ 5. | ~~(4.000)~~ 3.497 |

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

### SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.



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Insertion times taken from GE Document 22A 1342
"CONTROL ROD Drive System Design Spec."

See Spec. 3.1.3.1.b

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## REACTIVITY CONTROL SYSTEMS

## FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

| Position Inserted From Fully Withdrawn | | Average Scram Insertion Time (Seconds) | |
|---|---|---|---|
| ~~(46)~~ | 45 | ~~(0.398)~~ | 0.455 |
| ~~(36)~~ | 39 | ~~(1.169)~~ | 0.920 |
| ~~(26)~~ | 25 | ~~(2.120)~~ | 2.052 |
| ~~(6)~~ | 5 | ~~(4.300)~~ | 3.704 |

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

a. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

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

add

The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

2. With more than one control rod scram accumulator inoperable declare the associated control rods inoperable and:

a) If no Control Rod Drive hydraulic pumps are running and reactor pressure is less than 900 psig, immediately place the reactor mode switch in the Shutdown position.

b) Immediately verify control rod insertion capability by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range or place the reactor mode switch in the shutdown position.

c) Insert the inoperable control rods and disarm the associated directional control valves either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves

d) Otherwise, be in at least HOT SHUTDOWN within 12 hours.

Justification under 4.1.3.5.b; also, if the rods with inoperable accumulators can be inserted and meet all other requirements for being inoperable rods they are treated as such at other BWRs. Insertion, disabling and treating as inoperable is adequate. Being in HOT SHUTDOWN is unnecessary unless the requirements in 3/4.1.3 cannot be met.

## CONTROL ROD SCRAM ACCUMULATORS

### LIMITING CONDITION FOR OPERATION

3.1.3.5 ~~3.1.3.3~~ All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable:

   a) Within 8 hours:

      1) Restore the inoperable accumulator to OPERABLE status, or

      2) Declare the control rod associated with the inoperable accumulator inoperable.

   b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

delete and replace with

~~2. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:~~

~~a) Immediately verify control rod insertion capability by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range, or place the reactor mode switch in the Shutdown position.~~

~~b) Insert the inoperable control rods and disarm the associated control valves either:~~

~~1) Electrically, or~~

~~2) Hydraulically by closing the drive water and exhaust water isolation valves.~~

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod scram accumulator inoperable:

1. Insert the affected control rod and disarm the associated directional control valves within one hour, either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. The provisions of Specification 3.0.3 are not applicable.

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.



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Intent is to assure scram capability at design scram rate at all reactor pressures. New action statement (3.1.3.3.2.a) at reactor pressures < 900# ensures that stored energy in the accumulators plus whatever reactor pressure is present is available to scram the CRD's.

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## SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the pressure and leak detectors are not in the alarmed condition unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors,

with the alarm setpoint at greater than or equal to 940 psig on decreasing pressure.

delete

~~2. Verifying that the accumulator pressure and level remains above the alarm set point(s) for greater than or equal to 20 minutes with no control rod drive pump operating.~~

Oct 10 [illegible]

REACTIVITY CONTROL SYSTEMS

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism:

   1. Within 2 hours:

      a) If permitted by the RPCS, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

         1) Observing any indicated response of the nuclear instrumentation, and

         2) Demonstrating that the control rod will not go to the overtravel position.

      can not be

      b) If recoupling ~~is not~~ accomplished ~~on the first attempt~~ or, if not permitted by the RPCS, then until permitted by the RPCS declare the control rod inoperable, insert the control rod and disarm the associated directional control valves either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

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

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

   1. Either:

      a) Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

      b) If recoupling is not accomplished, insert the control rod and disarm the associated directional control valve either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

   2. The provisions of Specification 3.0.3 are not applicable.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

[illegible]

## SURVEILLANCE REQUIREMENTS

4.1.3.6 ~~4.1.3.4~~ A control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod does not go to the overtravel position;

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

3. The provisions of Specification 3.0.4 are not applicable.

[illegible]

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## CONTROL ROD POSITION INDICATION (~~Optional, solid-state RSCS~~)

### LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable:

1. Within one hour:

   a) Determine the position of the control rod by (an alternate method), or

   b) Move the control rod to a position with an OPERABLE position indicator, or

   c) When THERMAL POWER is within the low power setpoint of the RSCS:

      1) Declare the control rod inoperable, and

      2) Verify the position and bypassing of control rods with inoperable "Full-in" and/or "Full-out" position indicators by a second licensed operator or other technically qualified ~~member of the unit technical staff.~~ individual

   d) When THERMAL POWER is greater than the low power setpoint of the RSCS, declare the control rod inoperable, ~~insert the control rod and disarm the associated directional control valves either:~~ and follow ACTION statement delineated in Specification 3.1.3.1b

      ~~1) Electrically, or~~

      ~~2) Hydraulically by closing the drive water and exhaust water isolation valves.~~

insert

2. 3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod. The provisions of Specification 3.0.3 are not applicable.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.



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[illegible]

## SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

a. At least once per 24 hours that the position of each control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6.b.

50 [illegible]

## CONTROL ROD DRIVE HOUSING SUPPORT

### LIMITING CONDITION FOR OPERATION

3.1.3.8

~~3.1.3.6~~ The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

### SURVEILLANCE REQUIREMENTS

4.1.3.8

~~4.1.3.6~~ The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

[illegible]

## 3/4.1.4 CONTROL ROD PROGRAM CONTROLS

## ROD WORTH MINIMIZER

## LIMITING CONDITION FOR OPERATION

3.1.4.1 ~~Control rod shall not be moved, except by scram.~~ The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*, when THERMAL POWER is less than or equal to (20)% of RATED THERMAL POWER, the low power setpoint.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified ~~member of the unit technical staff~~ individual who is present at the reactor control console. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position. The provisions of technical specification 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.1.4.1.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper annunication of the selection error of at least one out-of-sequence control rod.

b. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying the control rod patterns and sequence input to the RWM computer is correctly loaded following any loading of the program into the computer.

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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ROD SEQUENCE CONTROL SYSTEM ~~(Optional, Banked Position Type)~~

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod sequence control system (RSCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to 20% RATED THERMAL POWER, ~~the low power setpoint.~~

ACTION:

a. With the RSCS inoperable:

1. Control rod withdrawal for reactor startup shall not begin.
2. ~~Control rod movement shall not be permitted, except by a scram.~~

b. With an inoperable control rod(s), OPERABLE control rod movement may continue by bypassing the inoperable control rod(s) in the RSCS provided that:

1. The position and bypassing of inoperable control rods is verified by a second licensed operator or other technically qualified ~~member of the unit technical staff~~ individual, and
2. There are not more than 3 inoperable control rods in any RSCS group.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. ~~Determining that the rod control and information system is indicated to be operative prior to:~~ a. Performance of the self-test prior to

1. Each reactor startup, and
2. Rod inhibit mode automatic initiation when reducing THERMAL POWER.

b. Attempting to select and move an inhibited control rod:

1. After withdrawal of the first insequence control rod for each reactor startup, and
2. Within one hour after rod inhibit mode automatic initiation when reducing THERMAL POWER.

*See Special Test Exception 3.10.2
#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.



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Design Basis of RBM is to prevent exceeding LHGR limits during a single rod withdrawal. Below 30% power, LHGR will not be near a limiting value. FSAR 7.7.1.8.4 states that RBM will be passed if normalizing APRM signal is less than 30% power.

## ROD BLOCK MONITOR

## LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to ~~20%~~ 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable, restore the inoperable RBM channel to OPERABLE status within 24 hours and verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN; otherwise, place the inoperble rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

## SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

No heat tracing in WNP-2.

WNP-2

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## REACTIVITY CONTROL SYSTEMS

## 3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

### LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:
   1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.
   2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:
   1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 30 days or insert all insertable control rods within the next hour.
   2. With the standby liquid control system inoperable, insert all insertable control rods within one hour.
   3. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;
   1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.
   2. The available volume of sodium pentaborate ~~solution is greater than or equal to ( ) gallons, and,~~ within the limits of Fig. 3.1.5-2,
   
   delete ~~3. The heat tracing circuit is OPERABLE by determining the temperature of the (pump suction piping) to be greater than or equal to (70)°F.~~

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

2327 10.81

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

   1. Starting both pumps and recirculating demineralized water to the test tank.

   2. Verifying the continuity of the explosive charge.

   3. Determining that the available weight of sodium pentaborate is greater than or equal to ~~(X)~~ 5500 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-~~X~~ 2 by chemical analysis.*

   4. 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.

c. At least once per 18 months during shutdown by:

   1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.

   2. Demonstrating that the minimum flow requirement of X41.2X gpm at a pressure of greater than or equal to X1220X psig is met.

   3. Demonstrating that the pump relief valve setpoint is greater than or equal to ~~(1560)~~ 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

   4. ~~X~~Demonstrating that ~~all heat traced piping~~ the piping between the storage tank and the pump suction ~~and the reactor vessel~~ is unblocked by ~~X~~pumping from the storage tank to the test tank~~X~~ and then draining and flushing the piping with demineralized water.

   5. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized.

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

~~**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.~~



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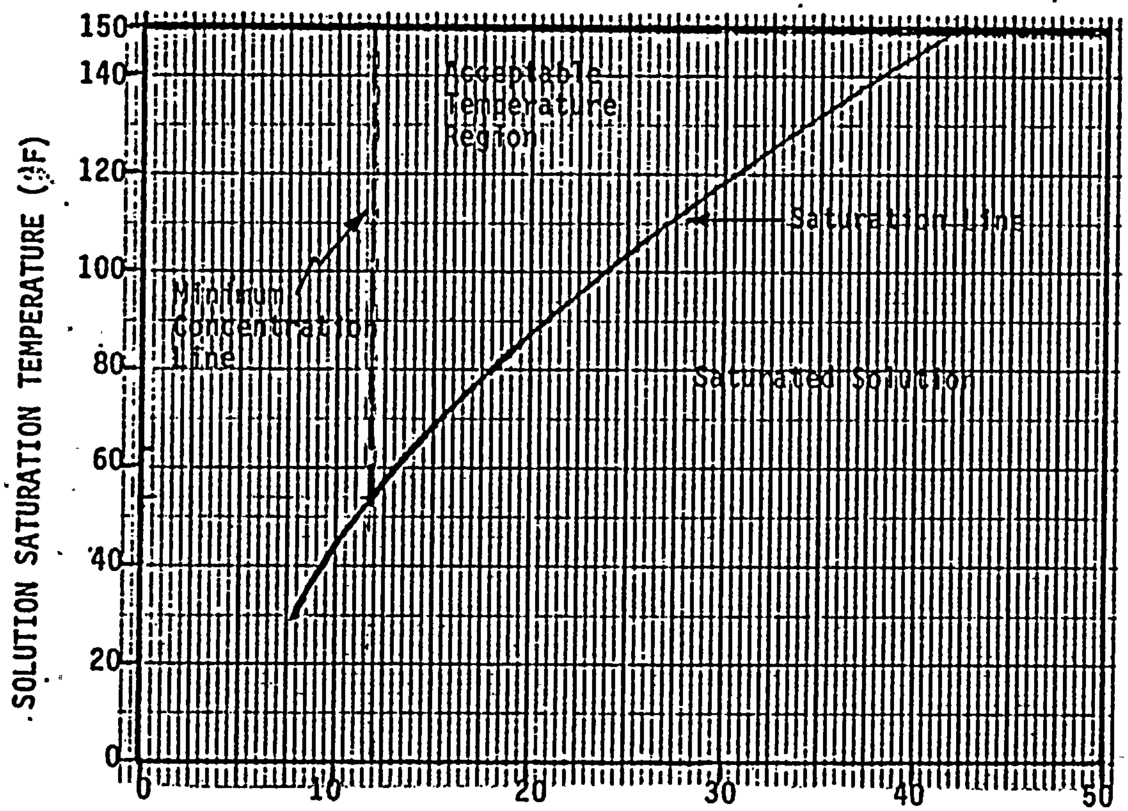
[illegible]




SOLUTION CONCENTRATION (% BY WEIGHT)

SODIUM PENTABORATE SOLUTION SATURATION TEMPERATURE

FIGURE 3.1.5-~~2~~ 1

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WNP-2

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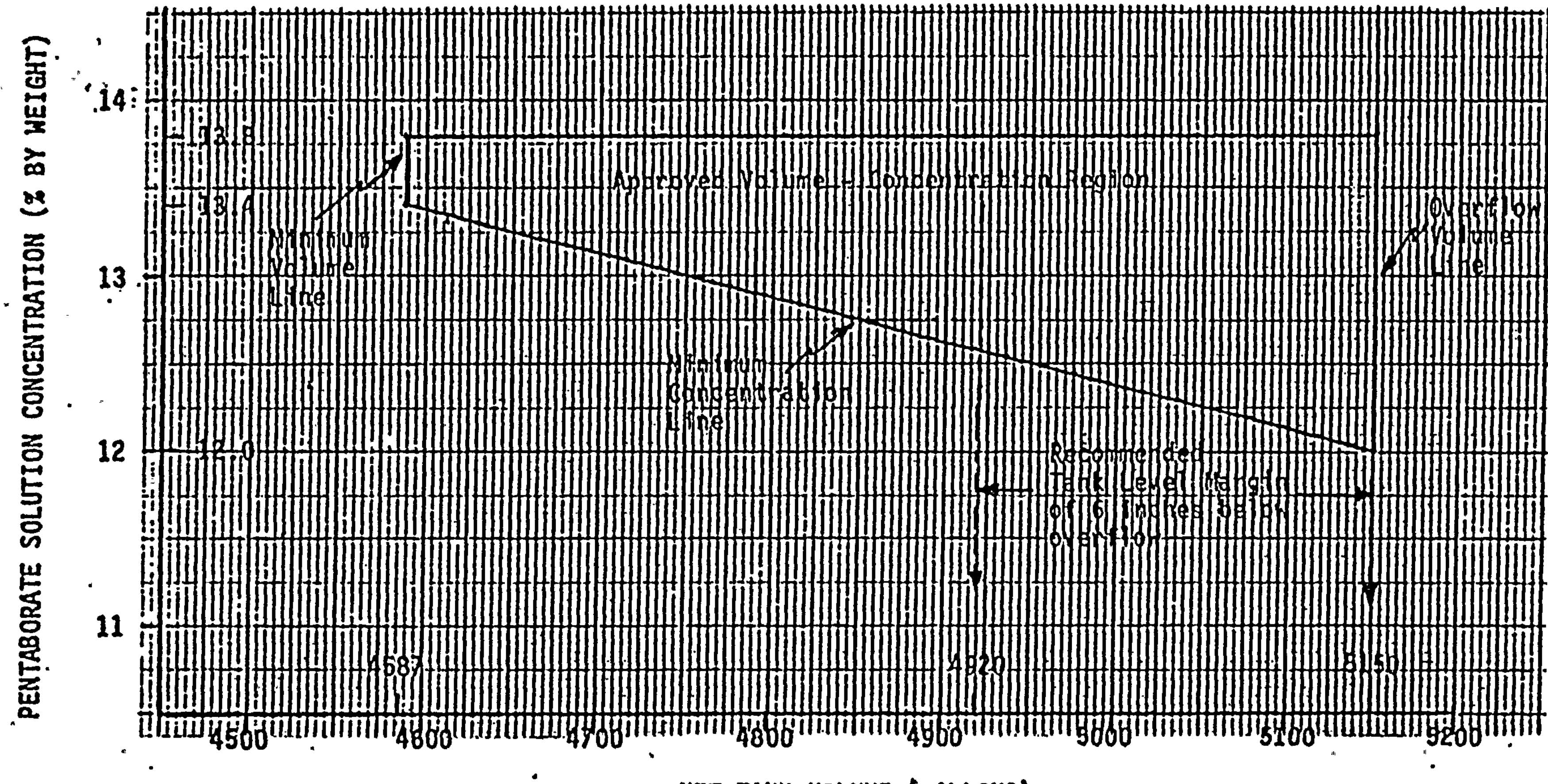


SODIUM PENTABORATE TANK, VOLUME VS. CONCENTRATION REQUIREMENTS
FIGURE 3.1.5-~~1~~ 2

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed ~~the limits shown in Figure 3.2.1-1, 3.2.1-2, and 3.2.1-3.~~ Tables 3.2.1-1; A, B and C

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of ~~Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3,~~ initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

#### SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from ~~Figure 3.2.1-1, 3.2.1-2, and 3.2.1-3:~~ Tables 3.2.1-1; A, B and C:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



FEB 01 1982

1. FSAR chapter 7, setpoint tables

2. NEDE 20944-P Sept A.2

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~~TABLE~~ 3.2.1-1.A

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

FUEL TYPE 8CR183

| EXPOSURE MWD/T | MAPLHGR KW/FT |
|---|---|
| 200 | 12.1 |
| 1000 | 12.2 |
| 5000 | 12.7 |
| 10000 | 12.9 |
| 15000 | 13.0 |
| 20000 | 12.7 |
| 25000 | 11.7 |
| 30000 | 10.8 |
| ~~35000~~ | ~~11.2~~ |

TABLE: 3.2.1-1B

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

FUEL TYPE 8CR233

| EXPOSURE MWD/T | MAPLHGR KW/FT |
|---|---|
| 200 | 12.1 |
| 1000 | 12.2 |
| 5000 | 12.4 |
| 10000 | 12.3 |
| 15000 | 12.3 |
| 20 000 | 12.2 |
| 25 000 | 11.7 |
| 30 000 | 11.4 |
| ~~25 000~~ | ~~11.4~~ |



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TABLE 3.2.1-1.C

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FUEL TYPE 8CR711

| EXPOSURE MWD/T | MAPLHGR KW/FT |
|---|---|
| 200 | 11.5 |
| 1000 | 11.4 |
| 5000 | 11.4 |
| 10000 | 11.5 |
| 15000 | 11.5 |
| 20000 | 11.0 |
| 25000 | 10.4 |
| 30000 | 9.7 |
| ~~35000~~ | ~~[illegible]~~ |

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## POWER DISTRIBUTION LIMITS

### 3/4.2.2 APRM SETPOINTS

### LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased simulated thermal power-upscale control rod block trip setpoint ($S_{RB}$) shall be established according to the following relationships:

$$S \leq (0.66W + (\cancel{54})\%)T \qquad 56$$
$$S_{RB} \leq (0.66W + (42)\%)T$$

where: S and $S_{RB}$ are in percent of RATED THERMAL POWER,
W = Loop recirculation flow in percent of rated flow,
T = Lowest value of the ratio of ~~(design TPF, (2.43) for (8 x 8) fuel, divided by the MTPF obtained for any class of fuel in the core)~~ FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased simulated thermal power-upscale control rod block trip setpoint less conservative than S or $S_{RB}$, as above determined, initiate corrective action within 15 minutes and restore S and/or $S_{RB}$ to within the required limits* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.2 The ~~(MTPF)~~ FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with ~~(MTPF)~~ MFLPD greater than or equal to ~~(2.43)~~ FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

INSERT

"c. With the main turbine bypass system INOPERABLE per Spec. 3.7.10, operation may continue and the provisions of specification 3.0.4 are not applicable, provided that, within one hour, MCPR, as a function of core flow, is determined to be greater than or equal to MCPR times the $K_f$ shown on Figure 3.2.3-1."

Reason for addition

The requirement for bypass system operability is analogous to the requirement for EOC-RPT system operability. With both EOC-RPT trip systems inoperable the Spec 3.3.4.2 allows ONE hour to return one trip system to operable status or take the action required by Specification 3.2.3. The action statements for these two analogous systems should be similar.

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO (Optional-ODYN Option A)

### LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than MCPR times the $K_f$ shown in Figure 3.2.3-1, (provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2), with MCPR for 8 x 8 fuel = ~~(1.20).~~ 1.24

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to ~~(~~25~~)~~% of RATED THERMAL POWER.

ACTION:

~~a.~~ With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR, as a function of core flow, is determined to be greater than or equal to MCPR times the $K_f$ shown in Figure 3.2.3-1, from:

1. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus ~~(2000)~~ MWD/t, with MCPR for ~~(~~8 x 8~~)~~ fuel = ~~(1.27)~~. Later
2. EOC minus ~~(2000)~~ MWD/t to EOC, with MCPR for 8x8 and 8x8R fuel = ~~(1.27)~~.

b. With MCPR, as a function of core flow, less than the applicable limit determined from Figure 3.2.3-1, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

INSERT

### SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, as a function of core flow, shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.



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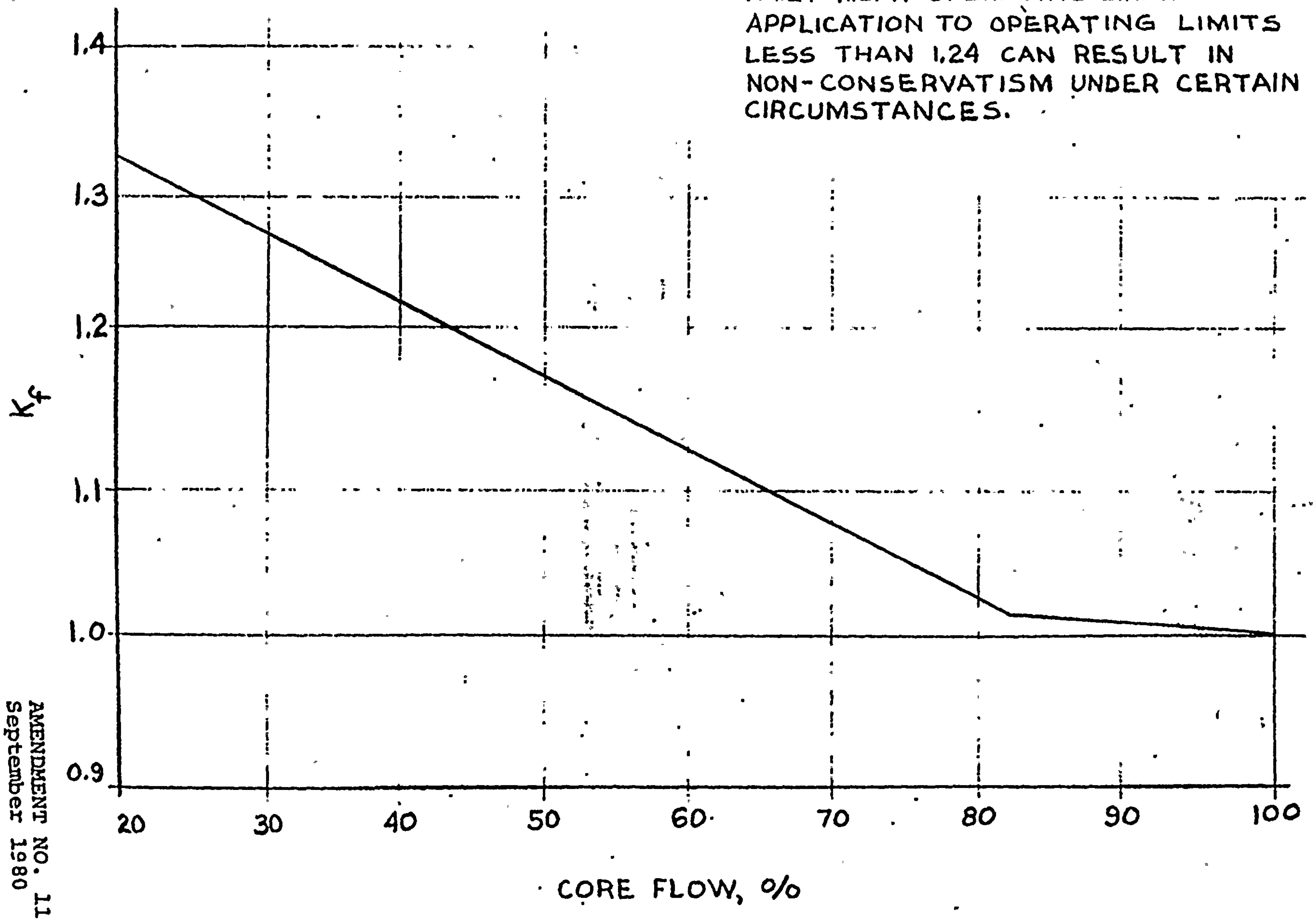
use this for figure 3.2.3-1

**NOTE:**
THIS CURVE WAS DEVELOPED BASED ON A 1.24 MCPR OPERATING LIMIT. APPLICATION TO OPERATING LIMITS LESS THAN 1.24 CAN RESULT IN NON-CONSERVATISM UNDER CERTAIN CIRCUMSTANCES.




WASHIN
WNP-2
SYSTEM

$K_f$ FACTOR VS. CORE FLOW

AMENDMENT NO. 11
September 1980

FIGURE 211.091-1

## 3/4.2.4 LINEAR HEAT GENERATION RATE

### LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kw/ft.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

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## 3/4.3 INSTRUMENTATION

### 3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

#### SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip ~~function~~ shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

functional unit

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**If both channels are inoperable in one trip system, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.



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SRM and IRM's provide adequate protection during cold shutdown (cond. 4).
so 15% upscale alarm not needed.

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## TABLE 3.3.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | APPLICABLE OPERATIONAL CONDITIONS | MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a) | ACTION |
|---|---|---|---|
| 1. Intermediate Range Monitors: | | | |
| a. Neutron Flux - High | 2 | 3 | 1 |
| | 3, 4 | 2 | 2 |
| | 5(b) | 3 | 3 |
| b. Inoperative | 2 | 3 | 1 |
| | 3, 4 | 2 | 2 |
| | 5 | 3 | 3 |
| 2. Average Power Range Monitor(c): | | | |
| a. Neutron Flux - Upscale (Setdown) | 2 delete | 2 | 1 |
| | 3, 4 | 2 | 2 |
| | 5(b) | 2 | 3 |
| b. Flow Biased Simulated Thermal Power - Upscale | 1 | 2 | 4 |
| c. Fixed Neutron Flux - Upscale | 1 | 2 | 4 |
| d. Inoperative | 1, 2 | 2 | 1 |
| | 3, 4 | 2 | 2 |
| | 5 | 2 | 3 |
| 3. Reactor Vessel Steam Dome Pressure - High | 1, 2(d) | 2 | 1 |
| 4. Reactor Vessel Water Level - Low, Level 3 | 1, 2 | 2 | 1 |
| 5. Main Steam Line Isolation Valve - Closure | 1(e) | 4 | 4 |
| 6. Main Steam Line Radiation - High | 1, 2(d) | 2 | 5 |

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | APPLICABLE OPERATIONAL CONDITIONS | MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a) | ACTION |
|---|---|---|---|
| 7. Primary Containment Pressure - High | 1, 2(f) | 2(g) | 1 |
| 8. Scram Discharge Volume Water Level - High | 1, 2 | 2 | 1 |
| | 5(h) | 2 | 3 |
| 9. Turbine ~~Stop~~ Throttle Valve - Closure | 1(i) | 4(j) | 6 |
| 10. Turbine ~~Control~~ Governor Valve Fast Closure, Valve Trip System Oil Pressure - Low | 1(i) | 2(j) | 6 |
| 11. Reactor Mode Switch Shutdown Position | 1, 2 | 1 | 1 |
| | 3, 4 | 1 | 7 |
| | 5 | 1 | 7 |
| 12. Manual Scram | 1, 2 | 2 | 1 |
| | 3, 4 | 2 | 8 |
| | 5 | 2 | 2 |

## TABLE 3.3.1-1 (Continued)

## REACTOR PROTECTION SYSTEM INSTRUMENTATION

### ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within ~~15 minutes and~~ reduce turbine first stage pressure to < ~~(250)~~ psig, equivalent 190 to THERMAL POWER less than ~~(30)~~% of RATED THERMAL POWER, within 25 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

## TABLE 3.3.1-1 (Continued)

## REACTOR PROTECTION SYSTEM INSTRUMENTATION

## TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than ~~(11)~~ 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed when turbine first stage pressure is < ~~(250)~~ 190 psig, equivalent to THERMAL POWER less than ~~(30)~~ 25% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

## TABLE 3.3.1-2

## REACTOR PROTECTION SYSTEM RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME (Seconds) |
|---|---|
| 1. Intermediate Range Monitors: | |
| a. Neutron Flux - High | NA |
| b. Inoperative | NA |
| 2. Average Power Range Monitor*: | |
| a. Neutron Flux - Upscale (Setdown) | NA |
| b. Flow Biased Simulated Thermal Power - Upscale | ≤ ~~(~~0.09~~)~~** |
| c. Fixed Neutron Flux - Upscale | ≤ ~~(~~0.09~~)~~ |
| d. Inoperative | NA |
| 3. Reactor Vessel Steam Dome Pressure - High | ≤ ~~(~~0.55~~)~~ |
| 4. Reactor Vessel Water Level - Low, Level 3 | ≤ ~~(~~1.05~~)~~ |
| 5. Main Steam Line Isolation Valve - Closure | ≤ ~~(~~0.06~~)~~ |
| 6. Main Steam Line Radiation - High | NA |
| 7. Primary Containment Pressure - High | NA |
| 8. Scram Discharge Volume Water Level - High | NA |
| 9. Turbine ~~Stop~~ throttle Valve - Closure | ≤ ~~(~~0.06~~)~~ |
| 10. Turbine ~~Control~~ governor Valve Fast Closure, Trip Oil Pressure - Low | ≤ ~~(~~0.08~~)~~# |
| 11. Reactor Mode Switch Shutdown Position | NA |
| 12. Manual Scram | NA |

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.



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5. channel functional test changed to Quarterly to be consistant with ASME quarterly valve testing requirements.

7. & 10. channel calibration frequency should be extended to 18 months (R) rather than Quarterly (Q) to be consistant with other protection system instruments. The frequency was originally R but was changed to Q due to since resolved instrument drift problems.

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TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL FUNCTIONAL TEST | CHANNEL CALIBRATION[a] | OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. Intermediate Range Monitors: | | | | |
| a. Neutron Flux - High | S/U[b] S | S/U[c], W | R | 2 |
| | S | W | R | 3, 4, 5 |
| b. Inoperative | NA | W | NA | 2, 3, 4, 5 |
| 2. Average Power Range Monitor[f]: | | | | |
| a. Neutron Flux - Upscale (Setdown) | S/U[b],S | S/U[c], W | SA | 2 |
| | S | W | SA | 3, 4, 5 |
| b. Flow Biased Simulated Thermal Power - Upscale | S | S/U[c], W | W[d][e], SA, R[h] | 1 |
| c. Fixed Neutron Flux - Upscale | S | S/U[c], W | W[d], SA | 1 |
| d. Inoperative | NA | W | NA | 1, 2, 3, 4, 5 |
| 3. Reactor Vessel Steam Dome Pressure - High | S | M | R[g] | 1, 2 |
| 4. Reactor Vessel Water Level - Low, Level 3 | S | M | R[g] | 1, 2 |
| 5. Main Steam Line Isolation Valve - Closure | NA | ~~M~~ Q | R[g] | 1 |
| 6. Main Steam Line Radiation - High | S | M | R | 1, 2 |
| 7. Primary Containment Pressure - High | NA | M | ~~[illegible]~~ R | 1, 2 |



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9. see explanation for changes in item 5.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL FUNCTIONAL TEST | CHANNEL CALIBRATION | OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 8. Scram Discharge Volume Water Level - High | (S) | M | R[g] | 1, 2, 5 |
| 9. Turbine Stop Valve - Closure | (S) | ~~M~~ Q | R[g] | 1 |
| 10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low | (S) | M | R[g] | 1 |
| 11. Reactor Mode Switch Shutdown Position | NA | R | NA | 1, 2, 3, 4, 5 |
| 12. Manual Scram | NA | M | NA | 1, 2, 3, 4, 5 |

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least ~~(1)~~ 0.5 decade~~s~~ during each startup and the IRM and APRM channels shall be determined to overlap for at least ~~(1)~~ 0.5 decade~~s~~ during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Calibrate trip unit at least once per 31 days.

add (h) On a refueling outage frequency verify that thermal power time constant is 6±1 sec.

## 3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**If both channels are inoperable in one trip system, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

## SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation function.

## TABLE 3.3.2-1

## ISOLATION ACTUATION INSTRUMENTATION

| TRIP FUNCTION | VALVE GROUPS OPERATED BY SIGNAL (a) | MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b) | APPLICABLE OPERATIONAL CONDITION | ACTION |
|---|---|---|---|---|
| 1. PRIMARY CONTAINMENT ISOLATION | | | | |
| a. Reactor Vessel Water Level | | | | |
| 1) Low, Level 3 | (2, 6, 8)[c] | 2 | 1, 2, 3 | 20 |
| 2) Low Low, Level 2 | (1, 3)[c] | 2 | 1, 2, 3 | 20 |
| b. Drywell Pressure - High | (2, 6)[c] | 2 | 1, 2, 3 | 20 |
| c. Main Steam Line | | | | |
| 1) Radiation - High | (1)[d] | 2 | 1, 2, 3 | 21 |
| | (7)[d] | 2 | 1, 2, 3 | 22 |
| 2) Pressure - Low | (1) | 2 | 1 | 23 |
| 3) Flow - High | (1) | 2/line[e] | 1, 2, 3 | 21 |
| d. Main Steam Line Tunnel Temperature - High | (1) | 2/line[e] | 1, 2, 3 | 21 |
| e. Main Steam Line Tunnel Δ Temperature - High | (1) | 2[e] | 1, 2, 3 | 21 |
| f. Condenser Vacuum - Low | (1) | 2 | 1, 2*, 3* | 21 |
| g. Manual Initiation | (1) | (2)/(group) | 1, 2, 3 | (24) |
| | (2, 3, 6, 7) | (1)/(group) | 1, 2, 3 | (26) |
| | (8) | (1)/(valve) | 1, 2, 3 | (26) |
| h. ____________ | | | | |
| 2. SECONDARY CONTAINMENT ISOLATION | | | | |
| a. ~~Plant~~ Exhaust Plenum Radiation - High | (6)[c][f] | 2[g] | 1, 2, 3, and ** | 25 |
| b. Drywell Pressure - High | (6)[c][f] | 2 | 1, 2, 3 | 25 |
| c. Reactor Vessel Water Level - Low, Level ~~(3)~~ | (6)[c][f] | 2 | 1, 2, 3, and # | 25 |
| d. Refueling Floor Exhaust Radiation - High | (6)[c][f] | 2[g] | 1, 2, 3, and ** | 25 |
| ~~e. Manual Initiation~~ | ~~(6)[f]~~ | ~~(1)/(group)~~ | ~~1, 2, 3~~ | ~~(26)~~ |
| | ~~(6)[f]~~ | ~~(1)/(group)~~ | ~~**~~ | ~~(25)~~ |

Later

a

LOW

Reactor Building Vent

later



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| TRIP Function | valve groups | minimum operable channels | Applicable operational condition | Action |
| --- | --- | --- | --- | --- |
| b. Heat exchanger Area Temperature - High | 2 | 1 | 1, 2, 3 | |
| c. Pump Area Temperature-High | 4 | 1 | 1, 2, 3 | 22 |
| d. Filter Demineralizer Area Temperature High | 6 | 1 | 1, 2, 3 | 22 |
| e. Heat exchanger outlet temperature High | 5 | Later | Later | 22 |
| f. Heat exchanger Area ventilation Δ Temperature High | | 1 | 1, 2, 3 | 22 |
| g. pump Area Ventilation Δ Temperature High | | 1 | 1, 2, 3 | 22 |
| h. Filter Demineralizer Area ventilation Δ Temperature. | | 1 | 1, 2, 3 | 22 |

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WIP-2

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

| TRIP FUNCTION | VALVE GROUPS OPERATED BY SIGNAL (a) | MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b) | APPLICABLE OPERATIONAL CONDITION | ACTION |
|---|---|---|---|---|
| 3. REACTOR WATER CLEANUP SYSTEM ISOLATION | | | | |
| a. Δ Flow - High | (3) later | (1) | 1, 2, 3 | 22 |
| add b. ~~Heat Exchanger/Pump Area Temperature - High~~ | (3) | (1) | 1, 2, 3 | 22 |
| c. ~~Heat Exchanger/Pump Area Ventilation Δ Temp. -~~ High | (3) | (1) | 1, 2, 3 | 22 |
| i → ~~d.~~ SLCS Initiation | (3)[h] | NA | 1, 2, 3 | 22 |
| j → ~~e.~~ Reactor Vessel Water Level - Low Low, Level 2 | (3) | 2 | 1, 2, 3 | 22 |
| ~~f. Manual Initiation~~ | ~~(3)~~ | ~~(1)~~ | ~~1, 2, 3~~ | ~~(26)~~ |
| ~~g.~~ ______ | ___ | (1)/(group) | ___ | ___ |
| 4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION | | | | |
| a. RCIC Steam Line Flow - High | (4) | (1) | 1, 2, 3 | 22 |
| b. RCIC Steam Supply Pressure - Low | (4)[i] Lake | 2 | 1, 2, 3 | 22 |
| c. RCIC Turbine Exhaust Diaphragm Pressure - High | (4) | 2 | 1, 2, 3 | 22 |
| d. RCIC Equipment Room Temperature - High | (4) | (1) | 1, 2, 3 | 22 |
| e. RCIC Steam Line Tunnel Temperature - High | (4) | (1) | 1, 2, 3 | 22 |
| f. RCIC Steam Line Tunnel Δ Temperature - High | (4) | (1) | 1, 2, 3 | 22 |
| X g. Drywell Pressure - High | (4) | (2) | 1, 2, 3 | (22) X |
| h. Manual Initiation | (4)[j] | (1)/(valve) | 1, 2, 3 | (26) |
| X ______ | - | - | - | - |

add "

| Trip Function | Valve Groups | Minimum Operable Channels | Applicable Operational Condition | Action |
| --- | --- | --- | --- | --- |
| b. RHR Equipment Area Ventilation Δ Temperature - High | | 1 | 1, 2, 3 | |
| c. RHR Equipment Area Temperature - High | | 1 | 1, 2, 3 | 27 |

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## TABLE 3.3.2-1 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION

| TRIP FUNCTION | VALVE GROUPS OPERATED BY SIGNAL (a) | MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b) | APPLICABLE OPERATIONAL CONDITION | ACTION |
|---|---|---|---|---|
| 5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION | | | | |
| add a. RHR Heat Exchanger steam Supply Flow - High | (5) - Later | — | 1, 2, 3 | 27 |
| ~~b. Manual Initiation~~ | ~~(5)~~ | ~~(1)/(valve)~~(j) | ~~1, 2, 3~~ | ~~(26)~~ |
| c. ____________ | - | - | - | - |
| 6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION | | | | |
| a. Reactor Vessel Water Level - Low, Level 3 | (5) later | 2 | 1, 2, 3 | 27 |
| b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High | (5) | 1 | 1, 2, 3 | 27 |
| c. RHR Equipment Area Δ Temperature - High | (5) | 1 | 1, 2, 3 | 27 |
| d. RHR ~~Area Cooler~~ Equipment Area Temperature - High | (5) | 1 | 1, 2, 3 | 27 |
| ~~e. Manual Initiation~~ | ~~(5)~~ | ~~(1)/(group)~~ | ~~1, 2, 3~~ | ~~(26)~~ |
| ~~f.~~ ____________ | | - | - | - |
| add e. Shutdown Cooling suction line high flow rate | | 1 | 1,2,3 | 27 |
| f. Heat exchanger Area Temperature - High | | 1 | 1,2,3 | 27 |
| g. Heat exchanger Area ventilation Δ Temperature | | 1 | 1,2,3 | 27 |

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## TABLE 3.3.2-1 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION

### ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

### NOTES

* May be bypassed with reactor steam pressure ≤ 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(c) Also actuates the standby gas treatment system.

(d) Also trips and isolates the mechanical vacuum pumps.

(e) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(f) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(g) One upscale and/or two downscale actuate the trip system.

(h) Closes only RWCU system inlet outboard valve.

(i) Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.

(j) Manual initiation isolates only and only with a coincident reactor vessel water level-low, level 3.

later

## TABLE 3.3.2-2

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| **1. PRIMARY CONTAINMENT ISOLATION** | | |
| a. Reactor Vessel Water Level | | |
| 1) Low, Level 3 | ≥ ~~(~~12.5~~)~~ inches* | ≥ ~~(~~11.0~~)~~ inches* |
| 2) Low Low, Level 2 | ≥ -~~(~~38~~)~~ inches* | ≥ -~~(~~45~~)~~ inches* |
| b. Drywell Pressure - High | ≤ ~~(~~ ~~)~~ psig 1.69 | ≤ ~~(~~ ~~)~~ psig 1.89 |
| c. Main Steam Line | | |
| 1) Radiation - High | ≤ (3.0) x full power background | ≤ (3.6) x full power background |
| 2) Pressure - Low | ≥ ~~(854)~~ psig 825 | ≥ ~~(834)~~ psig 805 |
| 3) Flow - High | ≤ ~~(140)% of rated flow~~ 104 psid | ≤ ~~(145)% of rated flow~~ 107 psid |
| d. Main Steam Line Tunnel Temperature - High | ≤ ~~(140)~~°F 145°F** | ≤ ~~( )°F~~ 150°F** |
| e. Main Steam Line Tunnel Δ Temperature - High | ≤ ~~(~~50~~)~~°F ** | ≤ ~~( )°F~~ 55°F** |
| f. Condenser Vacuum - Low | ≥ ~~(~~23~~)~~ inches Hg Absolute Pressure | ≥ ~~(~~ ~~)~~ inches Hg absolute pressure 24.5 |
| g. Manual Initiation | NA | NA |
| ~~h.~~ ________ | — | — |
| **2. SECONDARY CONTAINMENT ISOLATION** | | |
| a. ~~Plant~~ Reactor Building vent Exhaust Plenum Radiation - High | ≤ ~~(~~4.5~~)~~ mr/hr** | ≤ ~~(~~5.5~~)~~ mr/hr** |
| b. Drywell Pressure - High | ≤ ( ) psig 1.69 | ≤ ~~(~~ ~~)~~ psig 1.89 |
| c. Reactor Vessel Water Level - ~~Low, Level 3~~ Low Low Level 2 | ≥ ~~(12.5)~~ inches* -38" | ≥ ~~(11.0)~~ inches* -45 |
| d. Refueling Floor Exhaust Radiation - High | ≤ ~~(~~35~~)~~ mr/hr** | ≤ ~~(~~35~~)~~ mr/hr** |
| e. Manual Initiation | NA | NA |
| f. ________ | — | — |

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| Trip Function | Trip Setpoint | Allowable Value |
| --- | --- | --- |
| b. Heat Exchanger Area Temperature - High | ≤ 115°F ** | ≤ 120°F ** |
| c. Pump Area Temperaturer - High | ≤ 125° F ** | ≤ 130°F ** |
| d. Filter/Demineralizer Area Temperature-High | (Later) ** | (Later) ** |
| e Heat Exchanger outlet Temperature High | (Later) ** | (Later) ** |
| f Heat Exchanger Area Ventilation Δ Temperature High | ≤ 14°F ** | ≤ 20°F ** |
| g. Pump Area Ventilation Δ Temperature High | ≤ 50°F ** | ≤ 55°F ** |
| h. Filter Demineralizer Area Ventilation Δ Temperature - High | (Later) ** | (Later) ** |

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## TABLE 3.3.2-2 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| **3. REACTOR WATER CLEANUP SYSTEM ISOLATION** | | |
| a. Δ Flow - High | ≤ ~~(70)~~ gpm 57 | ≤ ~~(87.5)~~ gpm 76 |
| ~~b. Heat Exchanger/Pump Area Temperature - High~~ | ~~≤ (140)°F~~ | ~~≤ ( )°F~~ |
| add ~~c. Heat Exchanger/Pump Area Ventilation Δ Temperature - High~~ | ~~≤ (50)°F~~ | ~~≤ ( )°F~~ |
| c ~~d.~~ SLCS Initiation | NA | NA |
| d ~~e.~~ Reactor Vessel Water Level - Low Low, Level 2 | > (38) inches* | > (45) inches* |
| ~~f. Manual Initiation~~ | ~~NA~~ | ~~NA~~ |
| g. ____________ | ________ | ________ |
| **4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION** | | |
| a. RCIC Steam Line Flow - High | < ~~(300)% of rated flow~~ 6.5 psid | < ~~( )% of rated flow~~ 7.0 psid |
| b. RCIC Steam Supply Pressure - Low | ≥ ~~(60)~~ psig 57 | ≥ ~~(50)~~ psig 53 |
| c. RCIC Turbine Exhaust Diaphragm Pressure - High | ≤ ~~(~~10.0~~)~~ psig | ≤ ~~(~~20.0~~)~~ psig |
| d. RCIC Equipment Room Temperature - High | ~~≥ (175)°F ≤ (225)°F~~ ≤ 115 ** | ~~≥ ( )°F ≤ ( )°F~~ ≤ 120°F |
| e. RCIC Steam Line Tunnel Temperature - High | ≤ ~~(~~200~~)~~°F ** | ≤ ~~( )~~°F ** 205 |
| f. RCIC Steam Line Tunnel Δ Temperature - High | ≤ ~~(100)~~°F 120 ** | ≤ ~~( )~~°F 125 °F ** |
| ~~g.~~ Drywell Pressure - High | ≤ ~~( )~~ psig 1.69 | ≤ ~~( )~~ psig ** 1.89 |
| h. Manual Initiation | NA | NA |
| ~~i.~~ ____________ | ________ | ________ |



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| Trip Function | Trip Setpoint | Allowable Value |
|---|---|---|
| b. RHR equipment Area ventilation Δ Temperature High | later | Later |
| c. RHR Equipment Area Temperature - High | later | later |

add ①

add ②

| | | |
|---|---|---|
| RHR Pump A room | ≤ 62°F | ≤ 67°F |
| RHR Pump B room | ≤ 55 | ≤ 60°F |
| RHR pump C room | ≤ 50°F | ≤ 55°F |

add 3

| | | |
|---|---|---|
| e. shutdown cooling suction line high flow rate. | later | later |
| f. Heat exchanger area temperature - High | ≤ later | ≤ later |
| g. Heat exchanger area ventilation Δ Temperature - High | ≤ 55 °F | ≤ 60 °F |



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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
| --- | --- | --- |
| 5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION | | |
| a. RHR Heat Exchanger Steam Supply Flow - High | 6.8 psid $\leq$ ~~(300)% of rated flow~~ | 7.0 psid $\leq$ ~~( )% of rated flow~~ |
| ~~b. Manual Initiation~~ add 1 | ~~NA~~ | ~~NA~~ |
| ~~c.~~ ______ | ______ | ______ |
| 6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION | | |
| a. Reactor Vessel Water Level - Low, Level 3 | $\geq$ 12.5 inches* | $\geq$ ( ) inches* |
| b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High | $\leq$ (190) psig*** | $\leq$ ( ) psig*** |
| c. RHR Equipment Area $\Delta$ Temperature - High | $\leq$ (100)°F** | $\leq$ ( )°F** |
| add 2 | | |
| d. RHR ~~Area Cooler~~ Equipment Area Temperature - High | $\leq$ (200)°F** | $\leq$ ( )°F** |
| add | | |
| ~~e. Manual Initiation~~ | ~~NA~~ | ~~NA~~ |
| add 3 ~~f.~~ ______ | ___ | ___ |

*See Bases Figure B 3/4 3-1.
**Initial setpoint. Final setpoint to be determined during startup testing.
***Corrected for cold water head with reactor vessel flooded.



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| Trip Function | Response Time |
| --- | --- |
| b. Heat exchanger area temperature – High | NA |
| c. pump Area temperature - high | NA |
| d. Filter demineralizer area temperature – high | NA |
| e. Heat exchanger outlet temperature high | NA |
| f. heat exchanger area ventilation Δ Temperature - high | NA |
| g. pump area ventilation Δ Temperature high | NA |
| h. Filter demineralizer area Δ temperature – high | NA |

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## TABLE 3.3.2-3

## ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

| TRIP FUNCTION | RESPONSE TIME (Seconds)# |
|---|---|
| **1. PRIMARY CONTAINMENT ISOLATION** | |
| a. Reactor Vessel Water Level | |
| 1) Low, Level 3 | ≤ 13(a) |
| 2) Low Low, Level 2 | ≤ 1.0*/≤ 13(a)** |
| b. Drywell Pressure - High | ≤ 13(a) |
| c. Main Steam Line | |
| 1) Radiation - High(b) | ≤ 1.0*/≤ 13(a)** |
| 2) Pressure - Low | ≤ 1.0*/≤ 13(a)** |
| 3) Flow - High | ≤ 0.5*/≤ 13(a)** |
| d. Main Steam Line Tunnel Temperature - High | NA |
| e. Main Steam Line Tunnel Δ Temperature - High | NA |
| f. Condenser Vacuum - Low | NA |
| g. Manual Initiation | NA |
| ~~h.~~ ________ | ____ |
| **2. SECONDARY CONTAINMENT ISOLATION** | |
| a. ~~Plant~~ Reactor Building Vent Exhaust Plenum Radiation - High(b) | ≤ 13(a) |
| b. Drywell Pressure - High | ≤ 13(a) |
| c. Reactor Vessel Water Level - Low, ~~Level 3~~ Low, Level 2 | ≤ 13(a) |
| d. Refueling Floor Exhaust Radiation - High(b) super script | ≤ 13(a) |
| ~~e. Manual Initiation~~ | ~~NA~~ |
| ~~f.~~ ________ | ____ |
| **3. REACTOR WATER CLEANUP SYSTEM ISOLATION** | |
| a. Δ Flow - High | ≤ (13)(a) |
| ~~b. Heat Exchanger/Pump Area Temperature - High~~ | ~~(NA)~~ |
| ~~c. Heat Exchanger/Pump Area Ventilation Temperature ΔT - High~~ | ~~(NA)~~ |
| i. ~~d.~~ SLCS Initiation | NA |
| j. ~~e.~~ Reactor Vessel Water Level - Low Low, Level 2 | ≤ 13(a) |
| ~~f. Manual Initiation~~ | ~~NA~~ |
| g. ________ | ____ |

add



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add 1

Trip Function | Response Time

| Trip Function | Response Time |
|---|---|
| b. RHR Equipment Area ventilation Δ Temperature - High | NA |
| c. RHR Equipment Area Temperature-High | NA |

add 2

| | |
|---|---|
| e. shutdown cooling suction line high flow rate | Later |
| f. Heat exchanger Area temperature - High | NA |
| g. heat exchanger area ventilation Δ Temperature | NA |

## TABLE 3.3.2-3 (Continued)

## ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

| TRIP FUNCTION | RESPONSE TIME (Seconds)# |
|---|---|
| **4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION** | |
| a. RCIC Steam Line Flow - High | ≤ 13(a) |
| b. RCIC Steam Supply Pressure - Low | ≤ 13(a) |
| c. RCIC Turbine Exhaust Diaphragm Pressure - High | NA |
| d. RCIC Equipment Room Temperature - High | NA |
| e. RCIC Steam Line Tunnel Temperature - High | NA |
| f. RCIC Steam Line Tunnel Δ Temperature - High | NA |
| g. Drywell Pressure-High | NA |
| h. Manual Initiation | NA |
| f. ________ | ___ |
| **5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION** | |
| a. RHR Heat Exchanger Steam Supply Flow - High | < (13)(a) |
| ~~b. Manual Initiation~~ | ~~NA~~ |
| c. ________ | ___ |
| **6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION** | |
| a. Reactor Vessel Water Level - Low, Level 3 | ≤ 13(a) |
| b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High | NA |
| c. RHR Equipment Area Δ Temperature - High | NA |
| d. RHR ~~Area Cooler~~ equipment area Temperature - High | (NA) |
| ~~e. Manual Initiation~~ | ~~NA~~ |
| f. ________ | ___ |

add 1

add 2

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.



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## TABLE 4.3.2.1-1

## ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| TRIP FUNCTION | CHANNEL CHECK | CHANNEL FUNCTIONAL TEST | CHANNEL CALIBRATION | OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. PRIMARY CONTAINMENT ISOLATION | | | | |
| a. Reactor Vessel Water Level- | | | | |
| 1) Low, Level 3 | S | M | R | 1, 2, 3 |
| 2) Low Low, Level 2 | S | M | R | 1, 2, 3 |
| b. Drywell Pressure - High | (S) | M | (R) | 1, 2, 3 |
| c. Main Steam Line | | | | |
| 1) Radiation - High | S | M | R | 1, 2, 3 |
| 2) Pressure - Low | NA | M | Q | 1 |
| 3) Flow - High | S | M | R | 1, 2, 3 |
| d. Main Steam Line Tunnel Temperature - High | NA | M | Q | 1, 2, 3 |
| e. Main Steam Line Tunnel Δ Temperature - High | ~~NA~~ | M | Q | 1, 2, 3 |
| f. Condenser Vacuum - Low | (S) | M | (R) | 1, 2*, 3* |
| g. Manual Initiation | NA | M(a) | NA | 1, 2, 3 |
| h. ________ | — | — | — | ____ |
| 2. SECONDARY CONTAINMENT ISOLATION | | | | |
| a. Plant Exhaust Plenum Radiation - High | S | M | R | 1, 2, 3, and ** |
| b. Drywell Pressure - High | (S) | M | (R) | 1, 2, 3 |
| c. Reactor Vessel Water Level - Low, Level 3 | S | M | R | 1, 2, 3, and # |
| d. Refueling Floor Exhaust Radiation - High | S | M | R | 1, 2, 3, and ** |
| e. Manual Initiation | NA | M(a) | NA | 1, 2, 3, and ** |
| f. ________ | — | — | — | ____ |

| Trip Function | Channel Check | Channel Functional Test | Channel Calibration | Operational Conditions |
|---|---|---|---|---|
| b. Heat Exchanger Area Temperature - High | NA | M | Q | 1, [illegible] |
| c. Pump Area Temperature-High | NA | M | Q | 1, 2, 3 |
| d. Filter Demineralizer Area Temperature-High | NA | M | Q | 1, 2, 3. |
| e. Heat Exchanger Outlet Temperature - High | NA | M | Q | 1, 2, 3 |
| f. Heat Exchanger Area Ventilation Δ Temperature High | NA | M | Q | 1, 2, 3 |
| g. Pump Area Ventilation Δ Temperature High | NA | M | Q | 1, 2, 3 |
| h. Filter Demineralizer Area Ventilation Δ Temperature | NA | M | Q | 1, 2, 3 |

## TABLE 4.3.2.1-1 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

add

| TRIP FUNCTION | CHANNEL CHECK | CHANNEL FUNCTIONAL TEST | CHANNEL CALIBRATION | OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 3. REACTOR WATER CLEANUP SYSTEM ISOLATION | | | | |
| a. Δ Flow - High | S | M | R | 1, 2, 3 |
| ~~b. Heat Exchanger/Pump Area Temperature - High~~ | ~~NA~~ | ~~M~~ | ~~Q~~ | ~~1, 2, 3~~ |
| ~~c. Heat Exchanger/Pump Area Ventilation Δ Temperature - High~~ | ~~NA~~ | ~~M~~ | ~~Q~~ | ~~1, 2, 3~~ |
| i ~~d.~~ SLCS Initiation | NA | M(b) | NA | 1, 2, 3 |
| j ~~e.~~ Reactor Vessel Water Level - Low Low, Level 2 | S | M | R | 1, 2, 3 |
| ~~f. Manual Initiation~~ | ~~NA~~ | ~~M(a)~~ | ~~NA~~ | ~~1, 2, 3~~ |
| g. ________ | – | – | – | 1, 2, 3 |
| 4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION | | | | |
| a. RCIC Steam Line Flow - High | NA | M | Q | 1, 2, 3 |
| b. RCIC Steam Supply Pressure - Low | NA | M | Q | 1, 2, 3 |
| c. RCIC Turbine Exhaust Diaphragm Pressure - High | NA | M | Q | 1, 2, 3 |
| d. RCIC Equipment Room Temperature - High | NA | M | Q | 1, 2, 3 |
| e. RCIC Steam Line Tunnel Temperature - High | NA | M | Q | 1, 2, 3 |
| f. RCIC Steam Line Tunnel Δ Temperature - High | NA | M | Q | 1, 2, 3 |
| ~~g.~~ Drywell Pressure - High | ~~(S)~~ NA | M | ~~(R)~~ Q | 1, 2, 3~~X~~ |
| h. Manual Initiation | NA | ~~M(a)~~ R | NA | 1, 2, 3 |
| ~~i.~~ ________ | – | – | – | ________ |



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| Trip Function | Channel Check | Channel Functional Test | Channel Calibration | Operational Condition |
| --- | --- | --- | --- | --- |
| ① b. RHR equipment area ventilation Δ Temperature - High | NA | M | Q | 1,2,3 |
| c. RHR equipment area Temperature - High | NA | M | Q | 1, |

② e shutdown cooling suction line high flow rate

f Heat exchanger Area temperature - high

g heat exchanger area ventilation Δ Temperature - High

LATER

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| TRIP FUNCTION | CHANNEL CHECK | CHANNEL FUNCTIONAL TEST | CHANNEL CALIBRATION | OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION | | | | |
| add 1 | | | | |
| a. RHR Flow - High | NA | M(a) | Q | 1, 2, 3 |
| ~~b. Manual Initiation~~ | ~~NA~~ | ~~M~~ | ~~NA~~ | ~~1, 2, 3~~ |
| c. ______ | — | — | — | ______ |
| 6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION | | | | |
| add 2 | | | | |
| a. Reactor Vessel Water Level - Low, Level 3 | S | M | R | 1, 2, 3 |
| b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High | NA | M | Q | 1, 2, 3 |
| c. RHR Equipment Area Δ Temperature - High | NA | M | Q | 1, 2, 3 |
| d. RHR ~~Area Cooler~~ Equipment Area Temperature - High | NA | M(a) | Q | 1, 2, 3 |
| ~~e. Manual Initiation~~ | ~~NA~~ | ~~M~~ | ~~NA~~ | ~~1, 2, 3~~ |
| f. ______ | — | — | — | ______ |

* When reactor steam pressure > (1043) psig and/or any turbine stop valve is open.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATION and operations with a potential for draining the reactor vessel.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 31 days.

INSERT ① — " ... within
1. 7 days, provided that HPCS and RCIC systems are OPERABLE
2. 72 hours,
"
otherwise ..."

c. Revise "100" to "150" to agree with allowable value for shutdown cooling initiation (see 3.3.2-2 pg 3/4 3-17)

## 3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1. insert

c. With ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status ~~within 8 hours or~~ be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to ~~100~~ 150 psig within the following 24 hours.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

### SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS function.



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SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% ~~(99.95) (*)%~~ of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ~~(2300)~~ 4400 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than ~~(99.95) (*)%~~ 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 ~~(2300)~~ 4400 cfm ± 10%.

~~(*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99%, when a filter efficiency of 90% is assumed.)~~

add

d. at least once per 36 months by:

1. Verifying during a recombiner system functional test that upon introduction of 1% by volume hydrogen in a 140 to 180 SCFM stream containing at least 1% by volume oxygen that the catalyst bed temperature rises in excess of 120°F within a 20 minute period.