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SCHWENCER,A. Licensing Branch 2

Draft Tech Specs

see RPC

SUBJECT: Forwards Rev 2 of marked-up draft Tech Specs App A,Sections 1,2,3,4,5 & 6.Util review has been incorporated to ensure that rev addresses Unit 2 uniquely & meets requirements of STS,NUREG-0123.

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

April 6, 1983
G02-83-312
NS-L-02-PLP-83-021

Docket No. 50-397

Director of Nuclear Reactor Regulation
Attention: 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
DRAFT TECHNICAL SPECIFICATIONS,
REVISION 2, SUBMITTAL OF

Reference: Letter G02-82-130, G.D. Bouchey (SS) to
A. Schwencer (NRC), "Draft Technical
Specifications", dated February 1, 1982

Enclosed for your review are ten (10) copies of Revision 2 of the WNP-2 Technical Specifications. This revision is based on the initial submittal as forwarded by the referenced letter, NRC staff's comments on the initial submittal (provided by Mr. R. Bottimore in August 1982), and subsequent revisions provided by Mr. Bottimore.

An intensive WNP-2 staff review, incorporated in this submittal, has ensured that this revision addresses the WNP-2 plant uniquely, yet adequately meets the requirements of the Standard Technical Specifications, NUREG-0123.

With WNP-2 nearing construction completion the need for an approved set of plant specific Technical Specifications is evident. A June 1, 1983, approval is requested so that the Supply System staff will have sufficient time to complete the necessary surveillance procedures, change plant procedures to reflect the approved Technical Specifications, and prepare for licensing exams. In order to arrive at a mutually acceptable, coordinated program for finalizing the WNP-2 Technical Specifications the Supply System requests a meeting with your staff at their offices, April 21, 1983. Also enclosed is a proposed agenda.

B001
s 1/10
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Mr. A. Schwencer, Chief
Page Two
April 6, 1983
G02-83-312

Additionally, to facilitate staff review and avoid the time delay inherent in correspondence, telephone conference calls to clarify this submittal or arrange meetings can be coordinated through Messrs. P.L. Powell (509-377-2501 Ext. 2909) or R.M. Nelson, Manager, WNP-2 Licensing (509-377-2501 Ext. 2298).

Very truly yours,

AC Sorensen

for G.D. Bouchey
Manager, Nuclear Safety and Regulatory Programs

PLP/jca
Enclosures

cc:	R	Auluck	- NRC
	R	Bottimore	- NRC
	WS	Chin	- BPA
	A	Toth	- NRC Site
	D	Hoffman	- NRC

8304190551

WNP-2 TECHNICAL SPECIFICATION PROPOSED MEETING AGENDA

April 21-22, 1983
(As Necessary)

- WNP-2 Technical Specification Review, Requirements and Schedule
- WNP-2 Technical Specification Review Personnel
- NRC Technical Specification Approval Process, WNP-2 Plant Specific Schedule
- Discussion/Resolution of Technical Issues from Preliminary Review of the WNP-2 Technical Specifications
- Future Meetings to Support NRC WNP-2 Technical Specification Approval

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

(WASHINGTON NUCLEAR - UNIT 2)

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. ____

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Justification:

The additions to the definition section were made for the following reasons;

1. No definitions presently exist that clarify systems, tests or areas, extensively referred to in the environmental sections of these tech. spec's. The overriding intent is to preserve the bases for having a definition section.
2. "Protected area" was added due to the ever present confusion that exists between the protected area and restricted area.

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SECTION 1.0

DEFINITIONS

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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.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

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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.

1.8 Core Maximum Fraction of Limiting Power Density

CRITICAL POWER RATIO

1.~~8~~ 9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the (GEXL) correlations to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.~~9~~ 10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.~~10~~ 11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, 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.~~11~~ 12 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. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

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

1.~~12~~ 13 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~~ energization of the recirculation pump breaker trip coil ~~energization from initial movement of the associated:~~ from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine throttle valves channel sensor contact opening, and

b. Turbine governor valves initiation of valve fast closure.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

1.13 Changed as a result of LRG-1 issue #5 option.

DRAFT

DEFINITIONS

Exclusion/Site Area

FRACTION OF LIMITING POWER DENSITY

1.15 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

1.18 Gaseous Radwaste Treatment System

IDENTIFIED LEAKAGE

1.19 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, 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 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

1.20 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. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.21 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

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.23 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. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

1.24 Low Population Zone

GASEOUS RADWASTE TREATMENT SYSTEM

1.~~20~~ 18 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OFFSITE DOSE CALCULATION MANUAL

1.24. 27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

1.14 ~~Exclusion~~ site Area : A 1.2 mile radius area surrounding the reactor, in which the licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area (defined in 10CFR 100.3)

1.24 Low Population Zone : A 10 mile radius area immediately surrounding the exclusion area which contains residents, the total number and dencity of which are such that there is reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident (reference 10CFR 100.3).

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CORE, MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 ~~1.21~~ The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (cMFLPD) shall be highest value of the FLPD which exists in the core.

MAXIMUM TOTAL PEAKING FACTOR

1.22 25 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

1.23 26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel).

1.27 → Offsite Dose Calculation Manual

OPERABLE - OPERABILITY

1.24 1.28 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, electrical power, 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

1.25 1.29 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

1.26 1.30 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

1.27 1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.28 PRIMARY CONTAINMENT INTEGRITY shall exist when:
32

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

   1. Capable of being closed by an OPERABLE primary containment automatic isolation 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 in compliance with the requirements of 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 in compliance with the requirements of Specification 3.6.2.1.

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

1.33 → Process Control Program
1.34 → Protection Area

RATED THERMAL POWER

1.29 RATED THERMAL POWER shall be a total reactor core heat transfer rate to
35 the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.30 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from
36 when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE OCCURRENCE

1.31 A REPORTABLE OCCURRENCE shall be any of those conditions specified in
37 Specifications 6.9.1.8 and 6.9.1.9.

→ Restricted Area

ROD DENSITY

1.32 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.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the sampling, analysis,
.33 and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

1.34 Protected Area : The area defined by the WNP-2 security fence to which access is controlled.

1.38

Restricted Area : An area encompassing approximately one square mile has been established as the limit of the restricted area for which effluent concentrations have been calculated (reference 10CFR20.101) ... for the purposes of protecting personnel from exposure to radiation and radioactive materials. See FSAR section 2.1.1.3 for details. The shape of the area specifically excludes the WNP-4 parking lot and the BPA ASHE substation. It includes all within the WNP-2 protected area and the Emergency Operations Facility / Plant Support Facility and its access road.

DEFINITIONS

SECONDARY CONTAINMENT INTEGRITY

1.33 SECONDARY CONTAINMENT INTEGRITY shall exist when:
40
a. All secondary containment penetrations required to be closed during accident conditions are either:

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

2. Closed by at least one manual valve, blind flange, or deactivated automatic (valve) (or) (damper)(, as applicable,) 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 in compliance with the requirements of Specification 3.6.5.3.

d. (At least one) (The) door in each access to the secondary containment is closed (except for normal entry and exit).

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

(f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.)

SHUTDOWN MARGIN

1.34 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is
41 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.

1.42 → SOLIDIFICATION

→ SOURCE CHECK 1.43

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:
44
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 THERMAL POWER shall be the total reactor core heat transfer rate to the
45 reactor coolant.

SOLIDIFICATION

1.42 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

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

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TOTAL PEAKING FACTOR

1.37 46 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 47 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval ~~from when the (monitored parameter exceeds its actuation setpoint at the channel sensor) (turbine bypass control unit generates a turbine bypass valve flow signal) until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.~~

UNIDENTIFIED LEAKAGE

1.39 48 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

1.49 Unrestricted Area

1.50 Ventilation Exhaust Treatment System

Rewrite

Turbine Bypass System Response Time

1.47 The Turbine Bypass System Response Time consists of two components: a) Time from initial movement of the main turbine stop valve or control valve until 80% of turbine bypass capacity is established and b) the time from initial movement of turbine stop valve until initial movement of turbine bypass valve.

Justification: Rewrite modified to reflect the design and rewrite of LCO 3/4.7.9, Main Turbine Bypass System per GE.

VENTILATION EXHAUST TREATMENT SYSTEM

1. ~~34~~ 50 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

1.49 Unrestricted Area

All areas not contained within the Restricted Area boundary.

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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.
P	Prior to each radioactive release

Justification:
P frequency added to account for environmental tech. spec. sections.

DRAFT

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#,##,***	≤ 200°F
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°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.

##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 Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled provided that the one-rod-out interlock is OPERABLE.

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

Changes with respect to single recirculation loop operation are the result of an analyses performed by GE specific to WNP-2 (reference WNP-2 SER, section 4.4.9, page 4-28). Values denoted (later), will be provided when the analysis is concluded.

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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 and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06, for two recirculation loop operation or (later) for single loop operation 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 for two recirculation loop operation or (later) for single loop operation 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 and comply with the requirements of Specification 6.7.1.

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 and comply with the requirements of Specification 6.7.1.

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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 ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

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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.

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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	~~≤~~ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	~~≤~~ 15% of RATED THERMAL POWER	≤ 20 % of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	~~≤~~ 0.66 W+51% -.66ΔW(a), with a maximum of	≤ 0.66 W+54 % -.66ΔW(a) with a maximum of
2) High Flow Clamped	~~≤~~ 113.5% of RATED THERMAL POWER	≤ 115.5 % of RATED THERMAL POWER
c. Fixed Neutron Flux-High	≤ 110% of RATED THERMAL POWER	≤ 120 % of RATED THERMAL POWER
d. Inoperative	NA	NA
~~e. Downscale~~ Delete, N/A for WNP-2	~~≥ (5)% of RATED THERMAL POWER~~	~~≥ (3)% of RATED THERMAL POWER~~
3. Reactor Vessel Steam Dome Pressure - High	~~≤~~ ~~1043~~ 1037 psig	1057 ≤ ~~1063~~ psig
4. Reactor Vessel Water Level - Low, Level 3	~~≤~~ ~~12.5~~ 13 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} ~~(Drywell)~~ Pressure - High	~~≤~~ ~~1.69~~ 1.68 psig	1.88 ≤ ~~1.80~~ psig
8. Scram Discharge Volume Water Level - High	~~≤ (16)% of full scale~~ 529'6" ele.	529'6" ele. ~~≤ (39)% of full scale~~
9. Turbine Stop Valve - Closure	~~≤~~ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	~~≤~~ ~~(580)~~ 1250 psig	≥ (1000) psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

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

TABLE 2.2.1-1 (Continued)

(a) The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W).

ΔW is defined as the difference between indicated drive flow (in percent of rated flow) between two-loop and single-loop operation at the same core flow.

$\Delta W = 0$ for two-loop operation
$\Delta W =$ (Later) for single-loop operation

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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NOTE

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

2.1 SAFETY LIMITS

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BASES

2.0 INTRODUCTION

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 for two recirculation loop operation only and (later) for single-loop operation. 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 x 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 x 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.

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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.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

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 Amendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

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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[(a)]
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

(a) This quanity equals (later) for single loop operation.

* 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, and applies to operation with both recirculation loops in operation

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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 Medium enrichment - 1.039 Low enrichment - 1.030

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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 1971 Edition, including Addenda through Summer 1971, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure, 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Addenda through Winter 1971 for the reactor recirculation piping, which permits a maximum pressure transient of 125%, 1565 psig, of design pressure, 1250 psig for suction piping and 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.

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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~~ an allowance for instrument drift specifically allocated for each trip in the safety analyses. The trip setpoint and allowable values also contain additional margin for calibration and instrument accuracy.

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 been analyzed. The results of these analyses are in Section 15.4 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

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-High setpoint; 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-High setpoint, a time ~~constant~~ delay of [6] ±[1] seconds is introduced into 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. CMFLPD 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 control valve fast closure and turbine stop valve closure trip ~~is~~ are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint 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.

7. {Primary Containment} ~~(Drywell)~~ 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 trips.

or a loss of drywell cooling.

to minimize heat loads to equipment located within the primary containment

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LIMITING SAFETY SYSTEM SETTING

BASES

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. ~~The trip setpoint for each scram discharge volume is equivalent to a contained volume of ( ) gallons of water.~~ (over)

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. 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~~ where the turbine bypass valves {fail to} operate} and an RPT occurs.

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 increases that could result from fast closure of the turbine control valves due to load rejection coincident with or without ~~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 governor valve emergency trip fluid 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~~ slower 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 15.2.2 of the Final Safety Analysis Report.

11. Reactor Mode Switch Shutdown Position

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

12. Manual Scram

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

Add to 8.

There is approximately ~~24~~ 28 gallons of water between the rod block trip instrumentation ~~(per SDV)~~ at 527'1½" ele. and the scram level. The high level alarm only for SDV "A" is set at 525'2⅜" ele and the SDV "B" is set at 524'7 13/16" ele. The capacity between the high level alarms is approximately 6.4 gallons. The capacity between the SDV "A" high level and the rod block is approximately 50 gallons.

Justification: The information presented represents plant specific data and will aid the operator in evaluating the amount of leakage into the scram discharge volume.

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BASES FOR

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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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.

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.

This specification is not applicable in OPERATIONAL CONDITION 4 or 5.

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 or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

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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 Requirement.

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.

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 Specifications. Surveillance requirements do not have to be performed 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

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.

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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.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined 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.

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 the specified limit.

~~Initiate~~ Initiate a SHUTDOWN MARGIN determination

c. ~~Within~~ within one hour 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.

Spec. 4.1.1.c & 3.3.1.a.1.c requires that a SDM analysis to be performed (given a stuck rod) "within" one hour. The analysis will take approximately 6 hours (without considering occurrence on a back shift or other times when engineering support is not immediately available). The action statements for each requires hot shutdown in 6 hours and 12 hours respectively. Spec. 3/4.1.1 requires SDM be re-established in 6 hours but can only be accomplished after the actual SDM is determined. Given the flexibility/inconsistency between the 6 & 12 hour time limits and the restrictive analysis period of one hour, the spec (3/4.1.1) should require "initiation" of the analysis within one hour. The 6 hour limit would then become the time frame for determining or resolving the SDM anomoly.

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 equivalence difference exceeding 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.

DRAFT

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.

      b) Disarm the associated directional control valves** either:

         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      c) Comply with Surveillance Requirement 4.1.1.c.

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

   2. 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 trippable but inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within one hour:

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

      b) Demonstrate 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*.

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

      a) Electrically, or

      b) 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.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves** either:

   a) Electrically, or

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

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

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 by:

a. At least once per 31 days verifying each valve to be open,* and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the ~~(preset power level)~~ 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.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

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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 less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

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

2. Open when the scram signal is reset.

b. Proper float ~~(level sensor)~~ response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level ~~(if level measuring system)~~ instrumentation ~~(after each scram from a pressurized condition)~~ at least once per 31 days.

Delete

A CFT for the SDV scram float switches is required in the RPS LCO (3/4.3.1, table 4.3.3.1-1, item 8); & the Rod Block inst. LCO (3/4.3.6, table 4.3.6-1, item 5) covers the rod block float switch. There fore this addition is redundant and unnecessary.

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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:

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

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

2 ~~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.

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant 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 POWER OPERATION.

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

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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)
45	0.430
39	0.868
25	1.936
05	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.

DRAFT

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)
45	0.45~~5~~
39	0.92~~0~~
25	2.05~~2~~
5	3.70~~6~~

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

1. 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

2. 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.

b. The provisions of Specification 3.0.4 are not applicable.

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

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.

DRAFT

REACTIVITY CONTROL SYSTEMS

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 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, within 8 hours:

   a) Restore the inoperable accumulator to OPERABLE status, or

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

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

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

   a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one with-drawn control rod at least one notch 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.

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

3. The provisions of Specification 3.0.4 are not applicable.

b. In OPERATIONAL CONDITION 5*:

1. With one withdrawn control rod with its associated scram accumulator inoperable, 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. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

*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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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 indicated pressure greater than 940 is ~~(940) +(30), -(0)~~ psig 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, and verifying an alarm setpoint of [940] +[30], -[0] psig on decreasing pressure.

2. ~~(Verifying)~~ [Measuring and recording the time for up to 10 minutes] that [each individual] accumulator check valve maintains [the associated] accumulator pressure above the alarm set point ~~(for greater than or equal to 10 minutes)~~ with no control rod drive pump operating.

DRAFT

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

   1. If permitted by the RWM and RSCS, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

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

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

   2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM or RSCS, then until permitted by the RWM and RSCS, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or

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

   3. The provisions of Specification 3.0.4 are not applicable.

   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, either:

   1. 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

   2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or

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

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

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each affected 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 drive 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.

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CONTROL ROD POSITION INDICATION

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, within one hour:

1. Determine the position of the control rod by an alternate method, or

2. Move the control rod to a position with an OPERABLE position indicator, or

3. When THERMAL POWER is:

   a) Within the ~~(preset power level)~~ 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.

   b) Greater than the ~~(preset power level)~~ low power setpoint of the RSCS, 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.

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

4. The provisions of Specification 3.0.4 are not applicable.

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.

---

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

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

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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.

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CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 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 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.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 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 minimum allowable ~~(preset power level)~~ low power setpoint.

ACTION:

a. 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 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.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

b. In OPERATIONAL CONDITION 2 within 8 hours 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 demonstrating that the control rod patterns and sequence input to the RWM computer are 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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REACTIVITY CONTROL SYSTEMS

ROD SEQUENCE CONTROL SYSTEM

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 minimum allowable ~~(preset power level)~~ {low power setpoint}.

ACTION:

a. With the RSCS inoperable, 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, 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. Performance of a self-test:

   1. Within 8 hours prior to each reactor startup, and

   2. Prior to movement of a control rod after 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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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 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable:

1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable 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.

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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 otherwise 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 otherwise inoperable, insert all insertable control rods within one hour.

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 within the limits of Figure 3.1.5-2.

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

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

3. 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. Demonstrating that, when tested pursuant to Specification 4.0.5 ~~(at least once per 92 days)~~, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1220 psig is met.

d. 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~~ pump relief valve setpoint is ~~less~~ greater than or equal to 1400 psig but less than 1540 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

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

4. Demonstrating that the storage tank heaters are OPERABLE by verifying ~~a~~ the expected temperature rise of the sodium pentaborate solution in the storage tank ~~of at least ___°F within ___ minutes~~ 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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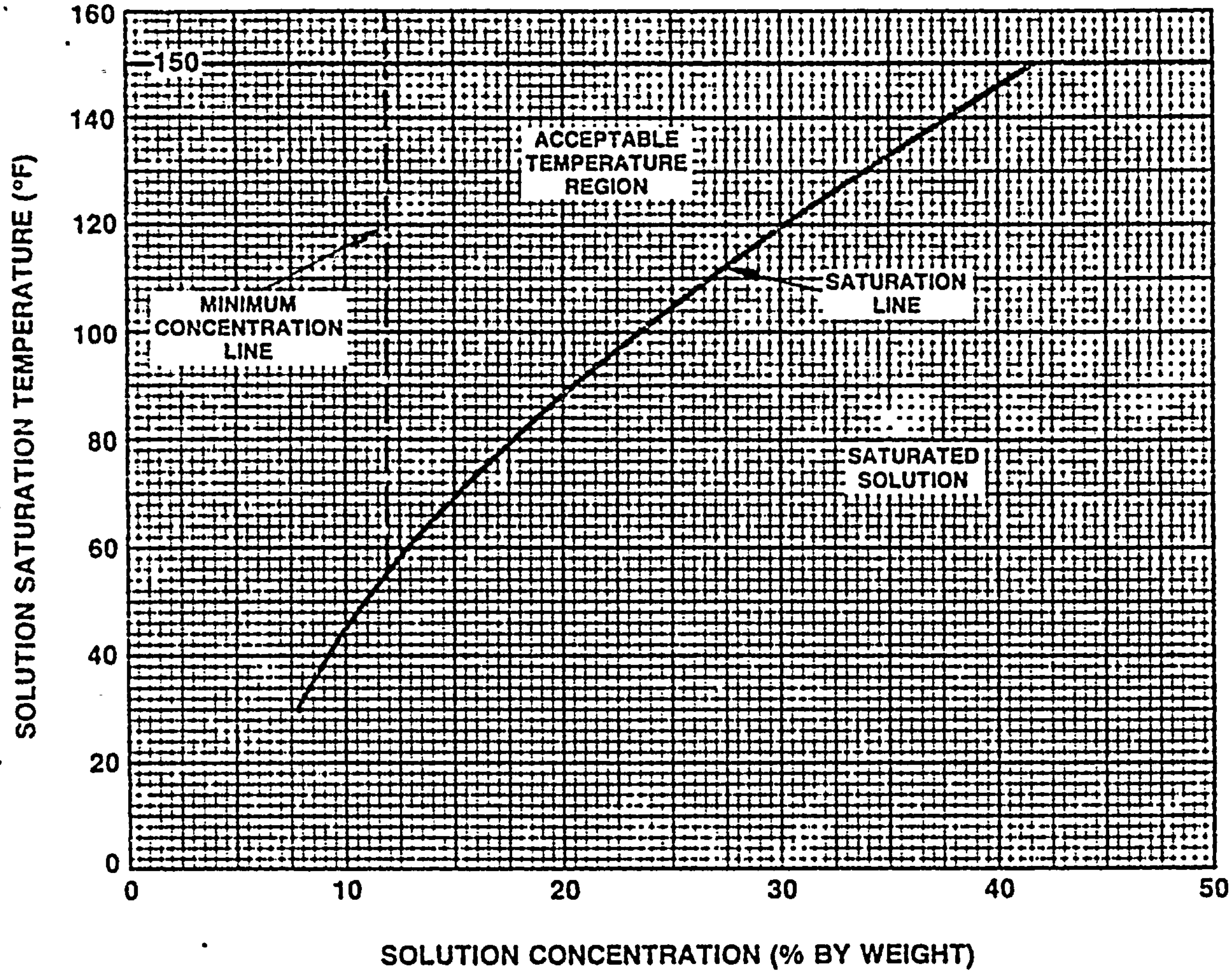
d.2) The safety relief valve setting is 1400 psig to insure an injection pressure of 1220 psig and protect the piping which is rated at 1540 psig. The change is intended to accurately reflect both concerns

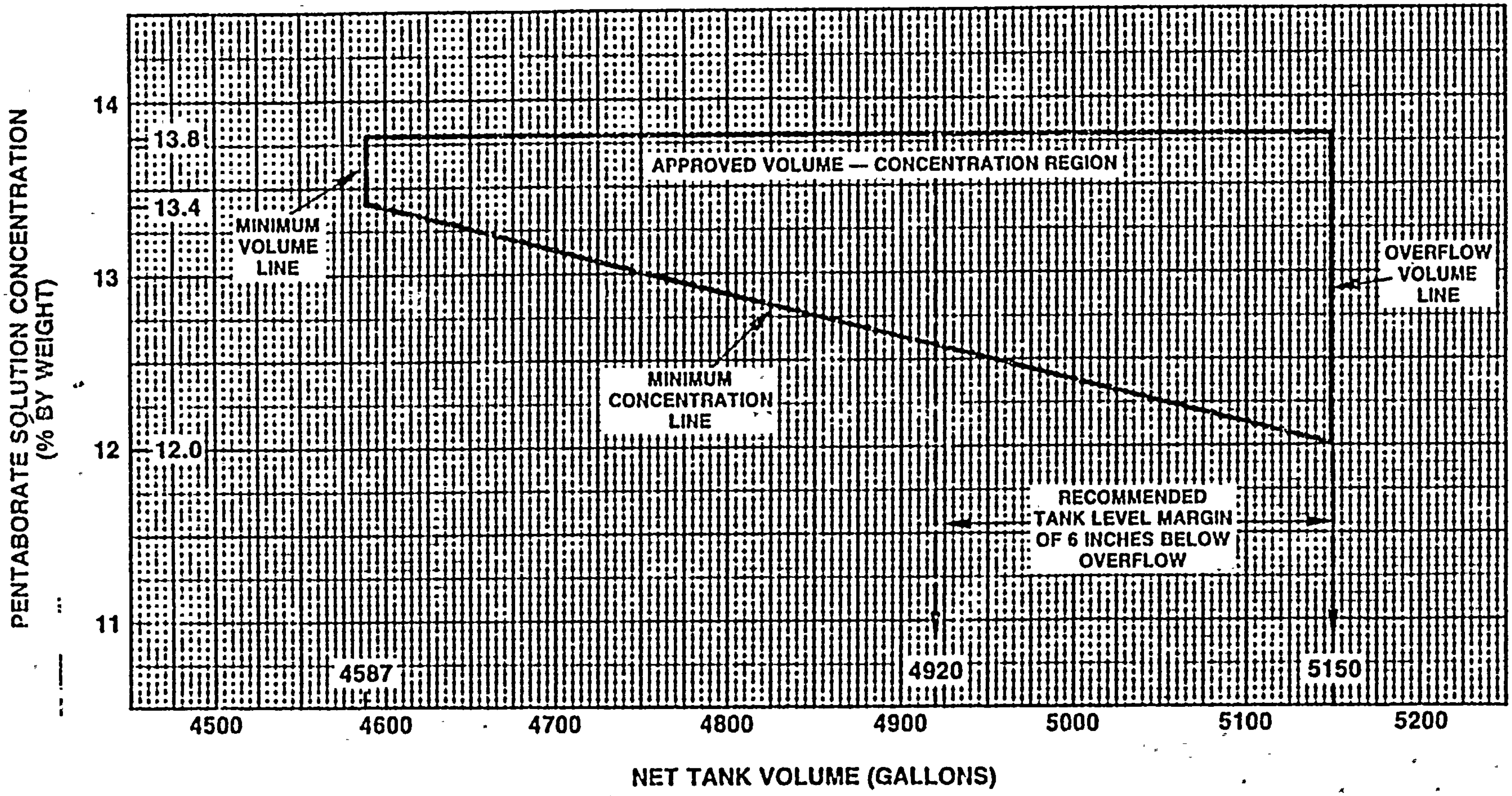
d.3) WNP-2 piping design does not employ heat tracing due to the suction valves proximity to the storage tank.

d.4) The heat up rate is determined during a startup test. To preclude completion of the test impacting tech. spec. submittal, the method employed by LaSalle has been adopted.



SODIUM PENTABORATE
SOLUTION SATURATION TEMPERATURE

FIGURE 3.1.5-1



SODIUM PENTABORATE TANK, VOLUME VS. CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-2

830323.1A

[illegible]

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 Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 for two recirculation loop operation. For single loop operation these values are reduced by multiplying by a factor of (later)

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 Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 and specification 3.2.1:

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 APLHGR.

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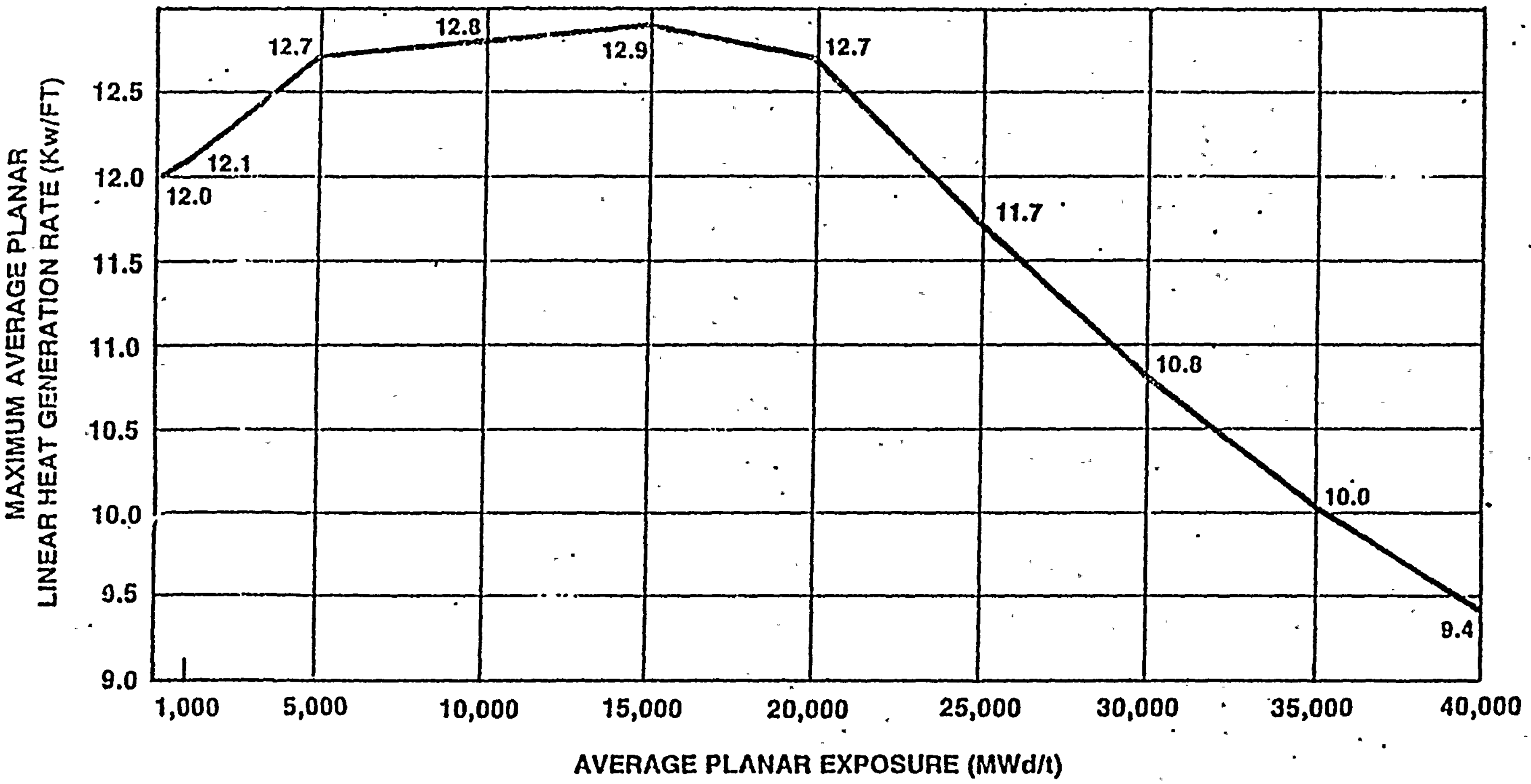
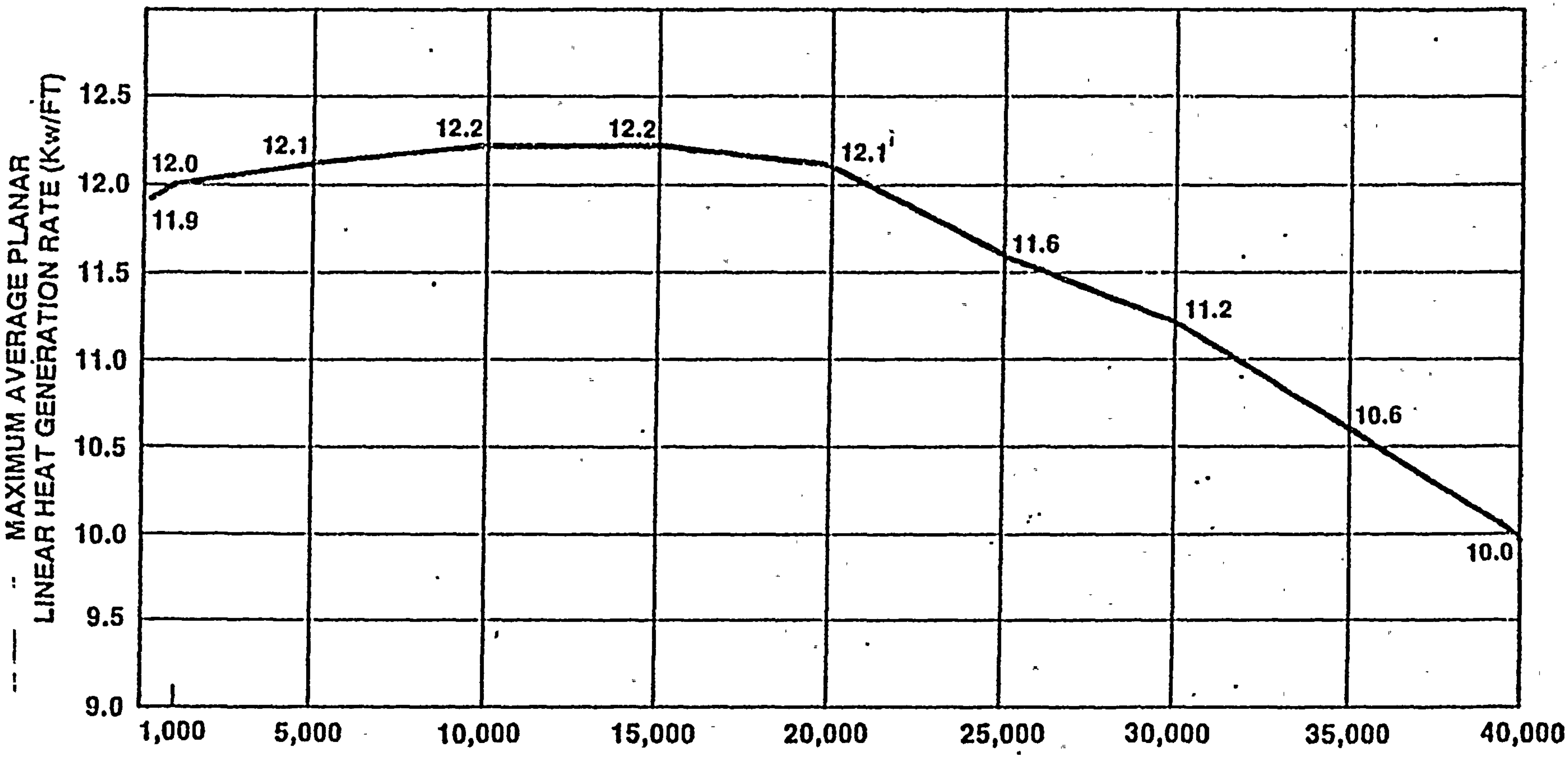
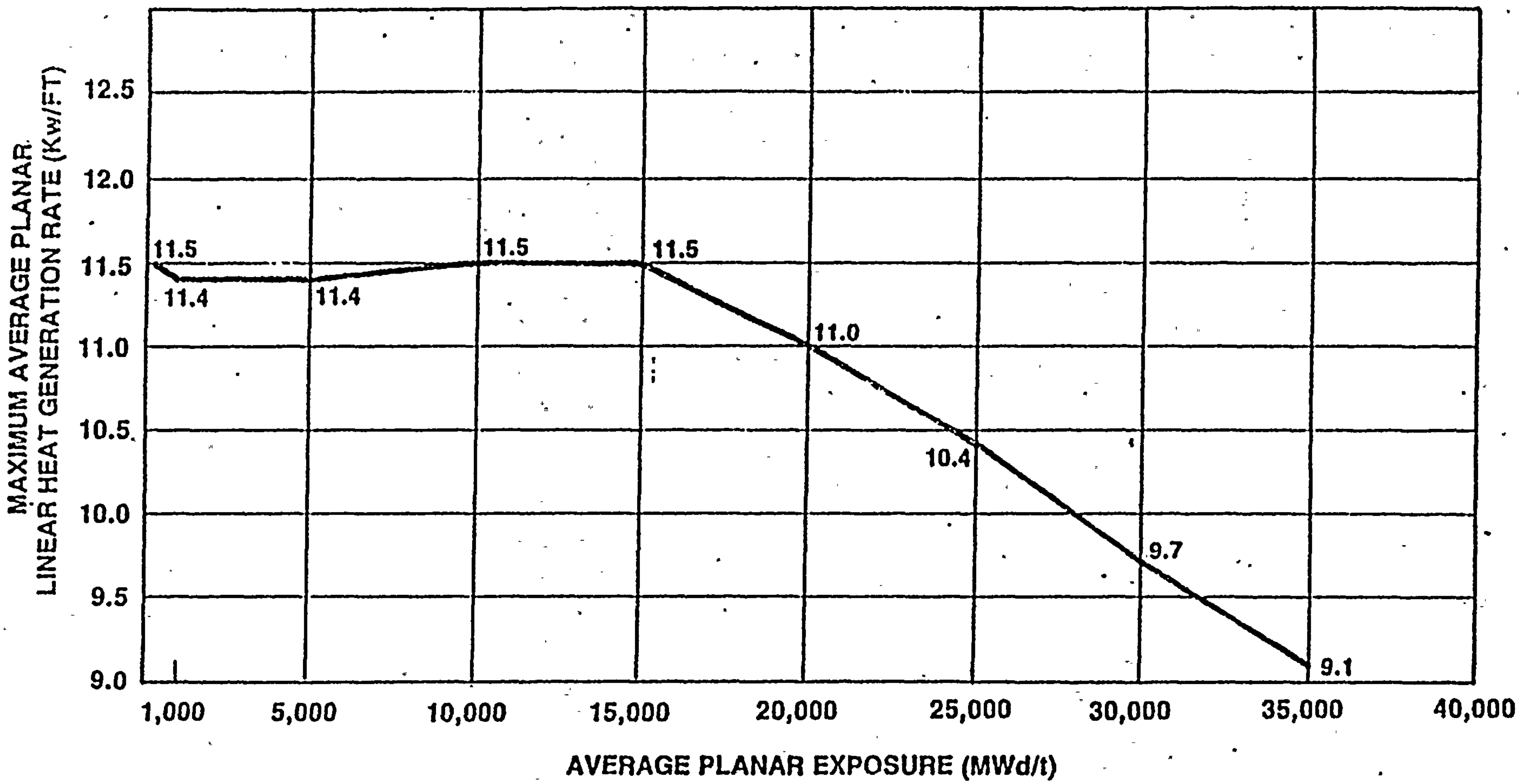


MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CR183
Figure 3.2.1-1



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CR233
Figure 3.2.1-2



MAXIMUM AVERAGE PLANAER LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE INITIAL CORE FUEL TYPE 8CR711

Figure 3.2.1-3

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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 neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66W' + 51\% - .66\Delta W)T$	$S \leq (0.66W + \{54\}\% - .66\Delta W)T$
$S_{RB} \leq (0.66W + \{42\}\% - .66\Delta W)T$	$S_{RB} \leq (0.66W + \{45\}\% - .66\Delta W)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

Insert following page info.

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of {108.5} million lbs/hr.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the CORE 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 neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value(*) 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 FRTP and the CMFLPD 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 flow biased neutron flux-upscale 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 CMFLPD greater than or equal to 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 CMFLPD, 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.

ΔW = Difference in indicated drive flow (in percent of rated drive flow) between two-loop and single-loop operation at the same core flow.

ΔW = 0 for two recirculation loop operation

= (Later) for single-loop operation

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit times the K_f determined from 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 limit for 8 x 8 fuel = 1.24 for two(2) recirculation loop operation. The MCPR limit is to be increased by (later) for single recirculation loop operation.

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

ACTION: Delete

~~(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 is determined to be equal to or greater than the MCPR limit times the K_f determined from 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).~~

~~2. EOC minus (2000) MWD/t to EOC, with MCPR for 8x8 and 8x8R fuel = (1.27).)~~

~~b.~~ With MCPR less than the MCPR limit times K_f 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.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the applicable MCPR limit times K_f, 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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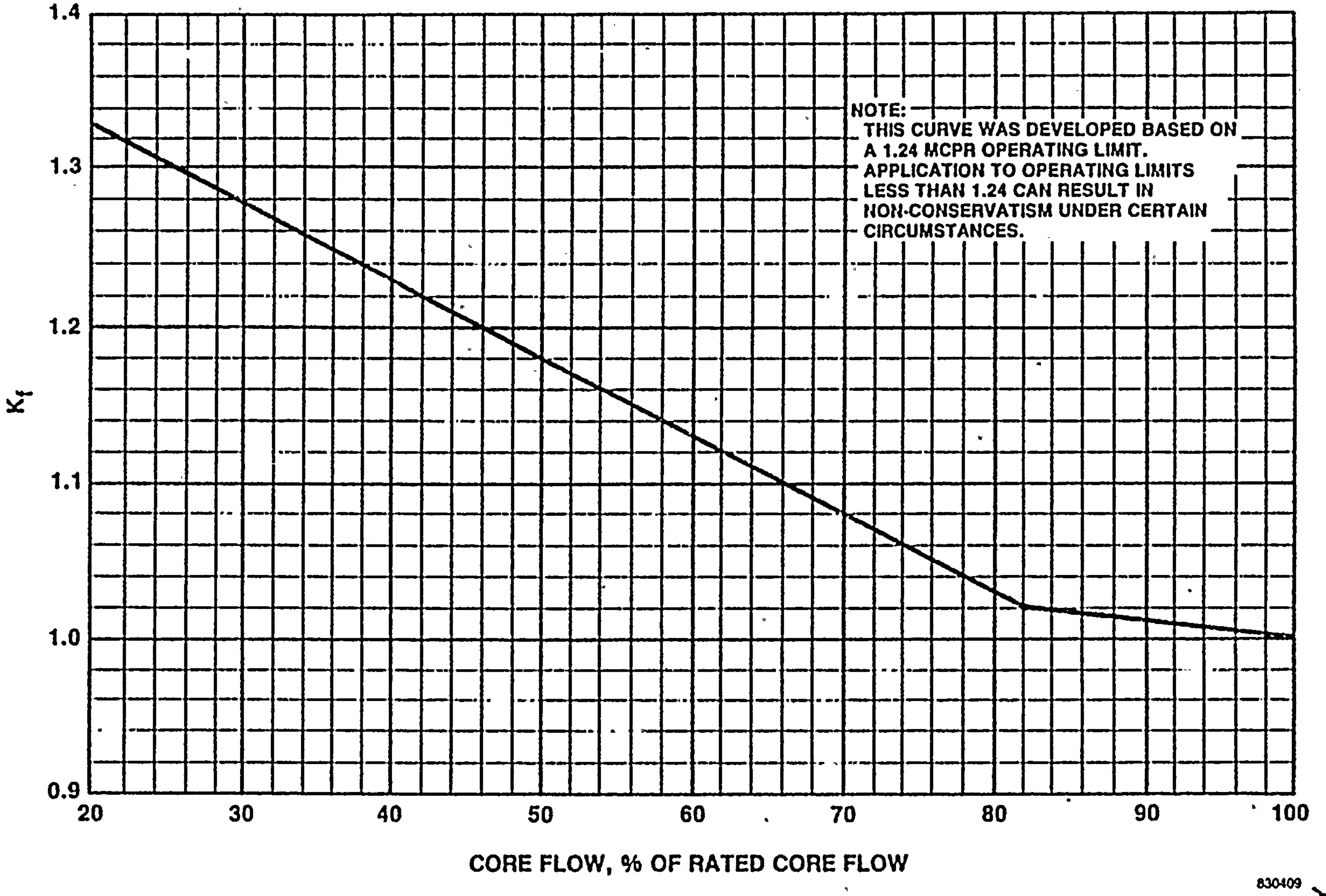
Changes from Standard

1) Standard LCO wording applies to G.E. BWR-6 product lines. Appropriate BWR-5 product line (WNP-2 plant specific) wording has been substituted.

2) The provision for EOC-RPT instrumentation operability was deleted because it was more correctly addressed by LCO 3/4. 3.4.2

3) A plant specific analysis has been completed for single recirc. loop operation which justifies the modification.

4) The action statement concerning EOC-RPT inoperability was deleted because it was more correctly addressed by LCO 3/4. 3.4.2



K_f FACTOR

FIGURE 3.2.3-1

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 the inoperable channel(s) and/or 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.

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 functional unit 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 channel per trip system 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 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 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

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, 4 5(b)	3 2 3	1 2 3
b. Inoperative	2 3, 4 5	3 2 3	1 2 3
2. Average Power Range Monitor(c):			
a. Neutron Flux - High, Setdown	2 3 5(b)	2 2 2	1 2 3
b. Flow Biased Simulated Thermal Power - High	1	2	4
c. Fixed Neutron Flux - High	1	2	4
d. Inoperative	1, 2 3 5	2 2 2	1 2 3
~~e. Downscale~~ Does not input to RPS	~~1(d)~~	~~2~~	~~4)~~
3. Reactor Vessel Steam Dome Pressure - High	1, 2(e)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(d)	4	4

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. Main Steam Line Radiation - High	1, $2^{(e)}$	2	5
7. Primary Containment ~~(Drywell)~~ Pressure - High	1, $2^{(f)}$	$2^{(g)}$	1
8. Scram Discharge Volume Water Level - High	1, 2 $5^{(h)}$	2 2	1 3
9. Turbine Throttle Valve - Closure	$1^{(i)}$	2-4$^{(j)}$ ← N/A at WNP-2	6
10. Turbine Governor Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(i)}$	$2^{(j)}$	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	~~2~~ 1 ~~2~~ 1 ~~2~~ 1	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

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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 < 180 psig, equivalent~~ ensure ~~to~~ THERMAL POWER is less than ~~25%~~ ≤30% of RATED THERMAL POWER, within 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.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and 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.

STEP 6:
REVISE PER GE INPUT. Also, the correlation between Reactor thermal power vrs. Turbine 1st stage pressure will be determined during the startup test program (power ascension testing program) and tech. spec will be finalized prior to that testing. The actual value will be incorporated into WNP-2 plant procedures and is unnecessary herein except that it will correspond to 30% of power.

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 if required by ~~circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations are being performed per~~ Specification ~~3.10.3.~~ 3.9.2.

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

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

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

(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 < 190 psig, equivalent to~~ THERMAL POWER is less than ~~25%~~ 30% 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.~~

Revised to remain consistent with proposed change to 3/4.9.2.

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(i) revise per GE input.

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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
~~(e. Downscale~~	~~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) ~~(Drywell)~~ Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Throttle Valve - Closure	≤ 0.06
10. Turbine 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, 6 ± 1 seconds.

#Measured from start of turbine control valve fast closure.

corrections per GE input

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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, [(b)]	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, [(b)]	S/U[(c)], W	SA	2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power - Upscale	S, D[(g)]	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, 5
~~(e. Downscale~~	~~S~~	~~W~~	~~SA~~	~~1)~~
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	~~M~~ Q	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2[(i)]
7. Primary Containment Pressure - High	~~S~~ NA	M	R	1, 2

Item 5 REVISE MSIV CLOSURE CHANNEL FUNCTIONAL TEST TO "Q" TO BE CONSISTENT WITH ASME QUARTERLY TESTING REQUIREMENTS AS CONDUCTED IN LCO 3/4.4.7.

Item 7 DRYWELL CHANNEL CHECK "NA" - THIS IS DONE ON A 12 HOUR BASIS FOR 3.6.1.6.

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 transmitter/trip unit	~~(S)~~ NA	M	~~(R)~~ Q	1, 2, 5$^{(j)}$
float switches	NA	Q	R	1, 2, 5 (j)
9. Turbine Throttle Valve - Closure	~~(S)~~ NA	M ~~M~~ ~~Q~~	~~(R)~~ Q	1
10. Turbine Governor Valve Fast Closure Valve Trip System Oil Pressure - Low	~~(S)~~ NA	M	~~(R)~~ Q	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 1/2 ~~decades~~ decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 ~~decades~~ decade 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% of RATED THERMAL POWER. 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) Verify measured core flow to be greater than or equal to established core flow at the existing flow control valve position.

(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

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

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

Item 8 correct table per design upgrade.

Channel check N/A as indication is not part of WNP-2 design upgrade nor it is required, therefore a CC cannot be performed.

Item 9 & 10
There is no indication in the control room that would enable a channel check to be performed.

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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 the inoperable channel(s) and/or 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.

*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 more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

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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 trip 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 channel per trip system 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 trip system.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	5g ~~(2, 6, 8)(b)~~	2	1, 2, 3	~~20~~ 22
2) Low Low, Level 2	(1, ~~3~~)(b) 2, 4	2	1, 2, 3	20
b. Drywell Pressure - High	4 ~~(2, 6)(b)~~	2	1, 2, 3	20
	5bg	2	1, 2, 3	22
c. Main Steam Line				
1) Radiation - High	(1)(c)	2	1, 2, 3	21
	2 ~~(7)~~(c)	2	1, 2, 3	22
2) Pressure - Low	(1)	2	1	23
3) Flow - High	(1)	2/line(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	(1)	~~2/line(d)~~	1, 2, 3	21
e. Main Steam Line Tunnel Δ Temperature - High	(1)	2 ~~(d)~~	1, 2, 3	21
f. Condenser Vacuum - Low	(1)	2	1, 2*, 3*	21
g. ~~Drywell and Suppression Chamber Radiation~~ Turbine Bldg Temperature - High	~~8~~ 1	2	1, 2, 3	~~22)~~ 21
h. Manual Initiation	(1)	(2)/(group)	1, 2, 3	~~(24)~~ 25
	~~(2, 3, 6, 7)~~	(1)/(group)	1, 2, 3	(25)(26)
	~~(8)~~ 5bg	(1)/(valve) group	1, 2, 3	(25)(26)
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	3 ~~(6)~~(b)(e)	2	1, 2, 3, and **	27
b. Drywell Pressure - High	3 ~~(6)~~(b)(e)	2	1, 2, 3	27
c. Reactor Vessel Water Level - Low Low, Level 2	3 ~~(6)~~(b)(e)	2	1, 2, 3, and #	27
~~d. Refueling Floor Exhaust Radiation - High~~	~~(6)(b)(e)~~	~~2~~	~~1, 2, 3, and **~~	~~27~~
d ~~e.~~ Manual Initiation	3b ~~(6)~~(e)	(1)/(group)	1, 2, 3	(25)(26)
	3b (6)(e)	(1)/(group)	**	(26)

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Table 3.3.2-1 Correct listings to be plant specific

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	7 ~~(3)~~	(1)	1, 2, 3	22
b. Heat Exchanger Area Temperature - High	7 ~~(3)~~	(1)	1, 2, 3	22
c. Heat Exchanger Area Ventilation Δ Temp. - High	7 ~~(3)~~	(1)	1, 2, 3	22
~~d. Heat Exchanger Outlet Temperature - High~~	~~( )~~	~~(1)~~	~~(1, 2, 3)~~	~~(22)~~
d ~~e~~. Pump Area Temperature - High	7 ~~( )~~	1	1, 2, 3	22
e ~~f~~. Pump Area Ventilation Δ Temp. - High	7 ~~( )~~	1	1, 2, 3	22
~~g. Filter/Demineralizer Area Temperature - High~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~22~~
~~h. Filter/Demineralizer Area Ventilation Δ Temp. - High~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~22~~
f ~~i~~. SLCS Initiation	7 ~~(3)~~ [illegible]	NA	1, 2, 3	22
g ~~j~~. Reactor Vessel Water Level - Low Low, Level 2	7 ~~(3)~~	2	1, 2, 3	22
h ~~k~~. Manual Initiation	7 (3)	(1)/group	1, 2, 3	(26)

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	~~(4)~~ 8	(1)	1, 2, 3	22
b. RCIC/RHR Steam Line Flow - High	8	1	1, 2, 3	22
c. ~~b.~~ RCIC Steam Supply Pressure - Low	~~(4)~~(h) 8, 9	2	1, 2, 3	22
d. ~~c.~~ RCIC Turbine Exhaust Diaphragm Pressure - High	~~(4)~~ 8	2	1, 2, 3	22
e. ~~d.~~ RCIC Equipment Room Temperature - High	~~(4)~~ 8	(1)	1, 2, 3	22
f. RCIC Equipment Room ΔTemp. High	8	1	1, 2, 3	22
g. ~~e.~~ RCIC Steam Line Tunnel Temperature - High	~~(4)~~ 8	(1)	1, 2, 3	22
h. ~~f.~~ RCIC Steam Line Tunnel Δ Temperature - High	~~(4)~~ 8	(1)	1, 2, 3	22
i. ~~g.~~ Drywell Pressure - High	~~(4)~~ 9	(2)	1, 2, 3	(22)
j. ~~h.~~ Manual Initiation	8 ~~(4)~~(h)	(1)~~/(valve)~~	1, 2, 3	(26)
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR ~~Heat Exchanger~~ Steam Line Supply Flow - High	~~(5)~~ 12	—	1, 2, 3	~~28~~ 22
b. Equipment Area Temperature - High	~~( )~~ 12	1	1, 2, 3	~~28~~ 22
c. Equipment Area Ventilation Δ Temp. - High	~~( )~~ 12	1	1, 2, 3	~~28~~ 22
d. ~~Manual Initiation~~	~~(5)(h)~~	~~(1)/(valve)~~	~~1, 2, 3~~	~~(26)~~
RCIC Steam Supply Pressure - Low	12	1	1, 2, 3	22

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	~~(5)~~ 6	2	1, 2, 3	28
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	~~(5)~~ 6	1	1, 2, 3	28
c. Equipment Area Temperature - High	~~(5)~~ 6	1	1, 2, 3	28
d. Equipment Area Ventilation Δ Temp. - High	~~(5)~~ 6	1	1, 2, 3	28
~~e. RHR Pump A Room~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~28~~
~~f. RHR Pump B Room~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~28~~
~~g. RHR Pump C Room~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~28~~
~~h. Heat Exchanger Area Temperature - High~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~28~~
~~i. Heat Exchanger Area Ventilation Δ Temp. - High~~	~~( )~~	~~1~~	~~1, 2, 3~~	~~28~~
e. ~~j.~~ Shutdown Cooling ~~Suction~~ Return Flow Rate - High	~~( )~~ 6	1	1, 2, 3	28
f ~~k.~~ Manual Initiation	~~(5)~~ 6	(1)/(group)	1, 2, 3	(26)

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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 - Restore the manual initiation function to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

ACTION 28 - 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~~ 1037 psig and all turbine stop valves ~~closed.~~ less than 90% open and keylock bypass switch in bypass position.

** 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) 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.

(b) Also actuates the standby gas treatment system.

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

(d) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.

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

(f) Closes only RWCU system isolation valve~~(s)~~ RWCU-V-1.

(g) ~~Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.~~ only valves RHR-V-123A and RHR-V-123B in valve Group 5 are required for primary isolation.

(h) Manual initiation isolates RCIC-V-8 only ~~and only with a coincident reactor vessel water level-low, level 3.~~ provided RCIC system is operating.

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Action 24 Deleted - Not used.

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	13.0 ≥ ~~12.5~~ inches*	11.0 ≥ ~~11.0~~ inches
2) Low Low, Level 2	-50 ≥ ~~-38~~ inches*	-57 ≥ ~~-45~~ inches
b. Drywell Pressure - High	1.68 ≤ ~~1.69~~ psig	1.88 ≤ ~~1.89~~ psig
c. Main Steam Line		
1) Radiation - High	3.0 ≤ ~~(2.5)~~ x full power background	3.6 ≤ ~~(3.0)~~ x full power background
2) Pressure - Low	831 ≥ ~~825~~ psig	811 ≥ ~~805~~ psig
3) Flow - High	105.5 ≤ ~~104~~ psid	108 ≤ ~~107~~ psid
d. Main Steam Line Tunnel Temperature - High	≤ ~~145°F~~** 150	200 ≤ ~~150°F~~**
e. Main Steam Line Tunnel Δ Temperature - High	≤ 50°F**	100 ≤ ~~55°F~~**
f. Condenser Vacuum - Low	≥ 23 inches Hg absolute pressure	≥ 24.5 inches Hg absolute pressure
g. Turbine Bldg Temperature - High	≤ 135 °F**	≤ 150 °F**
~~(g. Drywell and Suppression Chamber Radiation - High~~	~~≤ ( ) mR/hr**~~	~~≤ ( ) mR/hr**)~~
h. Manual Initiation	NA	NA
2. **SECONDARY CONTAINMENT ISOLATION**		
a. Reactor Building Vent Exhaust Plenum Radiation - High	5.0 ≤ ~~4.5~~ mR/hr**	11.6 ≤ ~~5.5~~ mR/hr**
b. Drywell Pressure - High	1.68 ≤ ~~1.69~~ psig	1.88 ≤ ~~1.89~~ psig
c. Reactor Vessel Water Level - Low Low, Level 2	-50 ≥ ~~-38~~ inches*	-57 ≥ ~~-45~~ inches
~~d. Refueling Floor Exhaust Radiation - High~~	~~≤ 35 mR/hr**~~	~~≤ 35 mR/hr**~~
d ~~e.~~ Manual Initiation	NA	NA

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Table 3.3.2-2 correct listings to be plant specific. Trip Setpoints & Allowable values revised per latest design information.

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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	≤ ~~67~~ 58.5 gpm	≤ ~~76~~ 65.5 gpm
b. Heat Exchanger Area Temperature - High	≤ ~~115~~ 135°F**	≤ ~~120~~ 200°F**
c. Heat Exchanger Area Ventilation Δ Temp. - High	≤ ~~14~~ 20°F**	≤ ~~20~~ 96°F**
~~d. Heat Exchanger Outlet Temperature - High~~	~~≤ ( )°F**~~	~~≤ ( )°F**~~
d ~~e~~. Pump Area Temperature - High	≤ ~~125~~ 130°F**	≤ ~~130~~ 150°F**
e ~~f~~. Pump Area Ventilation Δ Temp. - High	≤ 50°F**	≤ ~~55~~ 73°F**
~~g. Filter/Demineralizer Area Temperature - High~~	~~≤ ( )°F**~~	~~≤ ( )°F**~~
~~h. Filter/Demineralizer Area Ventilation Δ Temp. - High~~	~~≤ ( )°F**~~	~~≤ ( )°F**~~
f ~~i~~. SLCS Initiation	NA	NA
g ~~j~~. Reactor Vessel Water Level - Low Low, Level 2	≥ -~~38~~ 50 inches*	≥ -~~45~~ 57 inches
h ~~k~~. Manual Initiation	NA	NA

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ ~~6.5 psid~~ 290% Normal	≤ ~~7.0 psid~~ 300% Normal
b. RHR/RCIC Steam Line Flow - High	≤ 101.5" H_2O**	≤ 107.5" H_2O**
c ~~b~~. RCIC Steam Supply Pressure - Low	≥ ~~5~~ 62 psig	≥ ~~53~~ 58 psig
d ~~c~~. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ ~~10.0 psig~~ 174" H_2O**	≤ ~~20.0 psig~~ 183" H_2O**
e ~~d~~. RCIC Equipment Room Temperature - High	≤ ~~115~~ 130°F**	≤ ~~120~~ 200°F**
f. RCIC Equipment Room Δ Temp. - High	≤ 40°F**	≤ 123°F**
g ~~e~~. RCIC Steam Line Tunnel Temperature - High	≤ ~~200~~ 115°F**	≤ ~~205~~ 200°F**
h ~~f~~. RCIC Steam Line Tunnel Δ Temperature - High	≤ ~~120~~ 40°F**	≤ ~~125~~ 123°F**
i ~~g~~. Drywell Pressure - High	≤ 1.6~~9~~8 psig	≤ 1.8~~9~~8 psig
j ~~h~~. Manual Initiation	NA	NA
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION		
a. RHR ~~Heat Exchanger~~ Steam Line ~~Supply~~ Flow - High	≤ ~~6.0 psid~~ 101.5" H_2O**	≤ ~~7.0 psid~~ 107.5" H_2O**
b. Equipment Area Temperature - High (Pump Room A / Pump Room B)	≤ (136)°F** ≤ 132°F**	≤ (115)°F** ≤ 139°F**
c. Equipment Area Ventilation Δ Temp. - High (Pump Room A / Pump Room B)	≤ (63)°F** ≤ 56°F**	≤ (68)°F** ≤ 62°F**
d. ~~Manual Initiation~~ RCIC Steam Supply pressure - Low	~~NA~~ ≥60 psig	~~NA~~ ≥55 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 ~~12.5~~ inches*	≥ 11.0 ~~( )~~ inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 125 ~~(190)~~ psig#	≤ (135) psig#
c. Equipment Area Temperature - High Pump Room A	≤ 138 ~~(200)~~°F**	≤ (145)°F**
Pump Room B	≤ 132°F**	≤ 139°F**
d. Equipment Area Ventilation Δ Temp. - High Pump Room A	≤ 63 ~~(100)~~°F**	≤ (68)°F**
Pump Room B	≤ 56°F**	≤ 62°F**
~~e. Pump A Room~~	~~≤ 62°F~~	~~≤ 67°F~~
~~f. Pump B Room~~	~~≤ 55°F~~	~~≤ 60°F~~
~~g. Pump C Room~~	~~≤ 50°F~~	~~≤ 55°F~~
~~h. Heat Exchanger Area Temperature - High~~	~~≤ ( )°F~~	~~≤ ( )°F~~
e. ~~i.~~ Shutdown Cooling Return Flow Rate ~~Heat Exchanger Area Ventilation Δ Temp.~~ - High	~~≤ ( )°F~~ 174" H_2O**	~~≤ ( )°F~~ 183" H_2O**
f. ~~j.~~ Manual Initiation	NA	NA

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

**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

Reactor Vessel Dome pressure

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	≤ ~~10~~ 13 (a)
2) Low Low, Level 2	≤ 1.0*/≤ ~~10~~ 13 (a)**
b. Drywell Pressure - High	≤ ~~10~~ 13 (a)
c. Main Steam Line	
1) Radiation - High (b)	≤ 1.0*/≤ ~~10~~ 13 (a)**
2) Pressure - Low	≤ 1.0*/≤ ~~10~~ 13 (a)**
3) Flow - High	≤ 0.5*/≤ ~~10~~ 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. ~~Drywell and Suppression Chamber Radiation - High~~ Turbine Bldg. Temperature - High	NA
h. Manual Initiation	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Vent Exhaust Plenum Radiation - High (b)	≤ ~~10~~ 13 (a)
b. Drywell Pressure - High	≤ ~~10~~ 13 (a)
c. Reactor Vessel Water Level - Low Low, Level 2	≤ ~~10~~ 13 (a)
~~d. Refueling Floor Exhaust Radiation - High (b)~~	~~≤ 10 (a)~~
e. Manual Initiation	NA
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤ ~~10~~ 13 (a)(##)
b. Heat Exchanger Area Temperature - High	NA
c. Heat Exchanger Area Ventilation Δ Temp. - High	NA
~~d. Heat Exchanger Outlet Temperature - High~~	~~NA~~
d ~~e~~. Pump Area Temperature - High	NA
e ~~f~~. Pump Area Ventilation Δ Temp. - High	NA
~~g. Filter/Demineralizer Area Temperature - High~~	~~NA~~
~~h. Filter/Demineralizer Area Ventilation Δ Temp. - High~~	~~NA~~
f ~~i~~. SLCS Initiation	NA
g ~~j~~. Reactor Vessel Water Level - Low Low, Level 2	≤ ~~10~~ 13 (a)
h ~~k~~. Manual Initiation	NA

DEC 29 19[illegible]

Table 3.3.2-3 correct listings to be plant specific

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b. RHR/RCIC Steam Line Flow - High ≤13[a]

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 ~~10~~(a) ~~(###)~~
~~b.~~ c. RCIC Steam Supply Pressure - Low	≤ ~~10~~(a) ≤13
~~c.~~ d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
~~d.~~ e. RCIC Equipment Room Temperature - High	NA
~~e.~~ g. RCIC Steam Line Tunnel Temperature - High	NA
~~f.~~ h. RCIC Steam Line Tunnel Δ Temperature - High	NA
~~g.~~ i. Drywell Pressure-High	NA
~~h.~~ j. Manual Initiation	NA
f. RCIC Equipment Room Δ Temperature - High	NA
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	
a. RHR ~~Heat Exchanger~~ Steam Line ~~Supply~~ Flow - High	≤ 13 ~~10~~(a)
b. Equipment Area Temperature - High	NA
c. Equipment Area Ventilation Δ Temp. - High	NA
d. ~~Manual Initiation~~ RCIC Steam Supply Pressure - Low	NA
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. Equipment Area Temperature - High	NA
d. Equipment Area Ventilation Δ Temp. - High	NA
~~e. RHR Pump A Room~~	~~NA~~
~~f. RHR Pump B Room~~	~~NA~~
~~g. RHR Pump C Room~~	~~NA~~
~~h. Heat Exchanger Area Temperature - High~~	~~NA~~
~~i. Heat Exchanger Area Ventilation Δ Temp. - High~~	~~NA~~
e ~~j.~~ Shutdown Cooling Return ~~Suction~~ Flow Rate - High	NA
f ~~k.~~ Manual Initiation	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TABLE NOTATION

(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 diesel generator starting and sequence loading delays.

(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.

{##Time delay of {45} seconds} is effective during RWCU startup.

~~(##Time delay of ( ) seconds.)~~

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Time delay reference removed, (the 3 second time delay is included in the 13 sec.) Ref. RCIC DSDS 22A2869AT Rev.7.

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)~~ NA	M	(R)	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	M	R	1, 2, 3
2) Pressure - Low	(S)	M	(R)	1
3) Flow - High	S	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	(S)	M	(R)	1, 2, 3
e. Main Steam Line Tunnel Δ Temperature - High	(S)	M	(R)	1, 2, 3
f. Condenser Vacuum - Low	(S)	M	(R)	1, 2*, 3*
g. ~~Drywell and Suppression~~ Turbine Building Temperature ~~Chamber Radiation~~ - High	S	M	R	1, 2, 3
h. Manual Initiation	NA	~~(M(a))~~ (R)	NA	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	M	R	1, 2, 3, and **
b. Drywell Pressure - High	~~(S)~~ NA	M	(R)	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3, and #
~~d. Refueling Floor Exhaust Radiation - High~~	~~S~~	~~M(a)~~	~~R~~	~~1, 2, 3, and **~~
e. Manual Initiation	NA	~~(M(a))~~ (R)	NA	1, 2, 3, and **

Table 4.3.2.1-1 Correct Listings to be plant specific

1. b. } Drywell pressure - High Channel Check 'NA' - This is covered
2. b. } by section 4.6.1.6 Surveillance Requirement.

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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
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	(S)	M	(R)	1, 2, 3
c. Heat Exchanger Area Ventilation Δ Temperature - High	(S)	M	(R)	1, 2, 3
~~d. Heat Exchanger Outlet Temperature - High~~	~~NA~~	~~M~~	~~Q~~	~~1, 2, 3~~
d ~~e~~. Pump Area Temperature - High	~~NA~~ S	M	~~Q~~ R	1, 2, 3
e ~~f~~. Pump Area Ventilation Δ Temp. - High	~~NA~~ S	M	~~Q~~ R	1, 2, 3
~~g. Filter/Demineralizer Area Temperature - High~~	~~NA~~	~~M~~	~~Q~~	~~1, 2, 3~~
~~h. Filter/Demineralizer Area Ventilation Δ Temp. - High~~	~~NA~~	~~M~~	~~Q~~	~~1, 2, 3~~
f ~~i~~. SLCS Initiation	NA	M (R) a	NA	1, 2, 3
g ~~j~~. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
h ~~k~~. Manual Initiation	NA	~~(M(a))~~ (R)	NA	1, 2, 3

3e. & 3f. can be Channel Checked and are the same type as 3b. & 3c. which are checked by Channel Calibration on 'R'.

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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
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	(S)	M	(R)	1, 2, 3
b. RCIC/RHR Steam Line Flow - High	S	M	R	
c. ~~b.~~ RCIC Steam Supply Pressure - Low	(S)	M	(R)	1, 2, 3
d. ~~c.~~ RCIC Turbine Exhaust Diaphragm Pressure - High	(S)	M	(R)	1, 2, 3
e. ~~d.~~ RCIC Equipment Room Temperature - High	(S)	M	(R)	1, 2, 3
f. RCIC Eq. Room Δ Temp High	S	M	R	1 2 3
g. ~~e.~~ RCIC Steam Line Tunnel Temperature - High	(S)	M	(R)	1, 2, 3
h. ~~f.~~ RCIC Steam Line Tunnel Δ Temperature - High	(S)	M	(R)	1, 2, 3
i. ~~g.~~ Drywell Pressure - High	NA	M	~~Q~~ R	1, 2, 3
j. ~~h.~~ Manual Initiation	NA	~~M(a)~~ (R)	NA	1, 2, 3
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. ~~Heat Exchanger~~ RCIC/RHR Steam Supply Flow - High	(S)	M	(R)	1, 2, 3
b. Equipment Area Temperature - High	~~NA~~ S	M	~~Q~~ R	1, 2, 3
c. Equipment Area Ventilation Δ Temp. - High	~~NA~~ S	M	~~Q~~ R	1, 2, 3
d. ~~Manual Initiation~~	~~NA~~	~~M(a) (R)~~	~~NA~~	1, 2, 3
RCIC Steam Supply Pressure - Low	S	M	R	

5b.} Can be Channel Checked and are same type as 4e, f, g,
5c.} which are Channel Calibrated on 'R'.

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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
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	(S)ℓ	M	(R)ℓ	1, 2, 3
c. Equipment Area Temperature-High	(S)ℓ	M	(R)ℓ	1, 2, 3
d. Equipment Area Ventilation Δ Temp. - High	(S)ℓ	M	(R)ℓ	1, 2, 3
~~e. RHR Pump A Room~~	~~—~~	~~—~~	~~—~~	~~1, 2, 3~~
~~f. RHR Pump B Room~~	~~—~~	~~—~~	~~—~~	~~1, 2, 3~~
~~g. RHR Pump C Room~~	~~—~~	~~—~~	~~—~~	~~1, 2, 3~~
~~h. Heat Exchanger Area Temperature - High~~	~~—~~	~~—~~	~~—~~	~~1, 2, 3~~
~~i. Heat Exchanger Area Ventilation Δ Temp. - High~~	~~—~~	~~—~~	~~—~~	~~1, 2, 3~~
e. ~~j.~~ Shutdown Cooling ~~Suction~~ Return Flow Rate - High	NA	M	R	1, 2, 3
f. ~~k.~~ Manual Initiation	NA	~~(M(a))~~ (R)	NA	1, 2, 3

Surveillance not required when trip function bypassed (Reference: Table 3.3.2-1 note "#").

~~* 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.~~

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

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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.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.
2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to ~~350~~ psig within the following 24 hours.
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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 trip 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 channel per trip system 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 trip system.

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TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER ~~TRIP SYSTEM~~ FUNCTION (a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. LPCS Pump Discharge Flow-Low (Minimum Flow)	1	1, 2, 3, 4*, 5*	31
d. Reactor Vessel Pressure-Low (LPCS Permissive)	1	1, 2, 3, 4*, 5*	32 33
e. Reactor Vessel Pressure-Low (LPCI Permissive)	1	1, 2, 3, 4*, 5*	32 33
f. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
g. LPCI Pump A Discharge Flow-Low (Minimum Flow)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCS Pump Discharge Pressure-High (Pump Running)	(2)	1, 2, 3	32
f. LPCI Pump A Discharge Pressure-High (Pump Running)	(2)	1, 2, 3	32
g. Manual Initiation	2/division	1, 2, 3	35

① change "System" to "Function" in title: to avoid confusion with Trip System - applying to the system, Div. Trip system
MINIMUM OPERABLE CHANNELS PER TRIP <u>FUNCTION</u>
Function relating to the Functional units as Rx Vessel Water Lvl. etc.

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TABLE 3.3.3-1 (Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a) FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
B. DIVISION 2 TRIP SYSTEM			
1. RHR B &C (LPCI MODE)			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. Reactor Vessel Pressure-Low (LPCI Permissive)	1/valve	1, 2, 3, 4*, 5*	32 33
d. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
e. LPCI Pump Discharge Flow-Low (Minimum Flow)	1/pump	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3.	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCI Pump B and C Discharge Pressure - High (Pump Running)	(2)/(pump)	1, 2, 3	32
f. Manual Initiation	2/division	1, 2, 3	35

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TABLE 3.3.3-1 (Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP ~~SYSTEM~~ Function (a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	2	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. Reactor Vessel Water Level-High, Level 8	2[b]	1, 2, 3, 4*, 5*	32
d. Condensate Storage Tanks Level-Low	2[c]	1, 2, 3, 4*, 5*	36
e. Suppression Pool Water Level-High	2[c]	1, 2, 3, 4*, 5*	36
f. Pump Discharge Pressure-High (Pump Running)	(1)	1, 2, 3, 4*, 5*	32
g. HPCS System Flow Rate-Low (Minimum Flow)	(1)	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/~~system~~ division	1, 2, 3, 4*, 5*	34

TRIP FUNCTION	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. LOSS OF POWER					
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2/bus	1/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	3/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	38

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of 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.

* When the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

// Not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~150~~ 128 psig.

(b) Provides signal to close HPCS discharge valve only.

(c) Provides signal to HPCS pump suction valves only.

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels Function less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition within one hour* or declare the associated system inoperable.

b. For both trip systems, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels Function less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, declare the associated system inoperable. Function

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within one hour. Function

ACTION 34 - With the number of OPERABLE channels Function less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS valve or ECCS inoperable.

ACTION 35 With the number of OPERABLE channels Function less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the associated ECCS inoperable.

ACTION 36 With the number of OPERABLE channels Function less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable.

ACTION 37 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 38 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour*; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

*The provisions of Specification 3.0.4 are not applicable.

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TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ ~~1.69~~ psig 1.68	≤ ~~1.89~~ psig 1.88
c. LPCS Pump Discharge Flow-Low (minimum flow)	≥ ~~640~~ gpm 770	≥ ~~(520)~~ gpm 900
d. Reactor Vessel Pressure-Low (LPCS Permissive)	≥ ~~(696)~~ psig, 470 ~~(decreasing)~~	≥ ~~676~~ psig 454
e. Reactor Vessel Pressure-Low (LPCI Permissive)	≥ ~~(696)~~ psig, 470 (decreasing)	≥ ~~676~~ psig 454
f. LPCI Pump A Start Time Delay Relay	≤ 5 seconds	≤ 6 seconds
g. LPCI Pump A Discharge Flow-Low (minimum flow)	≥ ~~1000~~ gpm 1400	≥ ~~550~~ gpm 1250
h. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ ~~1.69~~ psig 1.68	≤ ~~1.89~~ psig 1.88
c. ADS Timer	≤ 105 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3 (Permissive)	≥ ~~12.5~~ inches* 13.0 14.5	≥ 11 inches 12.5 - 16.5
e. LPCS Pump Discharge Pressure-High (Pump Running)	~~≥~~ ~~146~~ psig, increasing	~~≥~~ ~~136~~ psig, increasing
f. LPCI Pump A Discharge Pressure-High (Pump Running)	~~≥~~ ~~119~~ psig, increasing 125	~~≥~~ ~~106~~ psig, increasing 115 - 135
g. Manual Initiation	NA	NA

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ ~~1.69~~ psig 1.68	≤ ~~1.89~~ psig 1.88
c. Reactor Vessel Pressure-Low (LPCI Permissive)	≥ ~~696~~ psig, ~~decreasing~~ 470	≥ 676 psig, ~~decreasing~~ 454
d. LPCI Pump B Start Time Delay Relay	≤ 5 seconds	≤ 6 seconds
e. LPCI Pump Discharge Flow-Low (minimum flow)	≥ 1000 gpm 1400	≥ 650 gpm 1250
f. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ ~~1.69~~ psig 1.68	≤ ~~1.89~~ psig 1.68
c. ADS Timer	≤ 105 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3 (Permissive)	≥ ~~12.5~~ inches* 13.0	≥ 11 inches
e. LPCI Pump B and C Discharge Pressure-High (Pump Running)	≥ ~~119~~ psig, ~~increasing~~ 125	≥ 106 psig, ~~increasing~~ 115-135
f. Manual Initiation	NA	NA

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches
b. Drywell Pressure - High	≤ ~~1.69~~ psig 1.68	≤ ~~1.89~~ psig 1.88
c. Reactor Vessel Water Level - High, Level 8	≤ 54.5 inches* 448'3"	≤ 56.0 inches 448'
d. Condensate Storage Tank Level - Low	≥ ~~(X+3)~~ inches ~~(**)~~ (#)	≥ ~~(X)~~ inches ~~(**)~~ (#)
e. Suppression Pool Water Level - High	≤ ~~(Y-3)~~ inches 466'8"	≤ ~~(Y)~~ inches 466'11"
f. Pump Discharge Pressure - High (Pump Running)	≥ ~~120~~ psig 140	≥ ~~110~~ psig 125
g. HPCS System Flow Rate - Low (Minimum Flow)	≥ ~~1000~~ gpm 1250	≥ ~~640~~ gpm 1200
h. Manual Intiation	NA	NA
D. LOSS OF POWER		
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage ###)	a. 4.16 kv Basis - 2870±(~~161~~ 86) volts b. 120 v Basis - 82±(~~5~~ 2.5) volts	2870±(~~315~~ 172) volts 82±(~~8~~ 5) volts
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - ~~3632±8.8~~ 1087 volts b. 120 v Basis - ~~103.8±0.25~~ 105 volts c. 8±~~0.04~~ .8 sec. time delay 3.0	~~3632±20.5~~ 216 volts ~~103.8±0.60~~ volts 8±0.8 sec. time delay 6.0

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

~~(**X is equivalent to 135,000 gallons of water in the condensate storage tank, a value at which the pump will not cavitate.)~~

~~(#Y is equivalent to 5 inches above normal water level of 466'-9-3/4".)~~

###These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

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TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 40
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	≤ 40
~~a. Pumps A and B~~	~~≤ 35~~
~~b. Pump C~~	~~≤ 40~~
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 27
5. LOSS OF POWER	NA

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TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	~~Q~~ R	1, 2, 3
c. LPCS Pump Discharge Flow-Low (minimum Flow)	NA	M	Q	1, 2, 3, 4*, 5*
d. Reactor Vessel Pressure-Low (LPCS Permissive)	NA	M	~~Q~~ R	1, 2, 3, 4*, 5*
e. Reactor Vessel Pressure-Low (LPCI Permissive)	NA	M	~~Q~~ R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Discharge Flow-Low (minimum Flow)	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	~~Q~~ R	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	S	M	R	1, 2, 3
e. LPCS Pump Discharge Pressure-High (Pump Running)	NA	M	Q	1, 2, 3
f. LPCI Pump A Discharge Pressure-High (Pump Running)	NA	M	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

Table 4.3.3.1-1

- Drywell pressure high - Revised Channel Calibration to remain consistant with RPS and Isolation Instr.

- Revise Channel Calibration on permissive signals to concure with trip system B and recently licensed plants (La Salle).

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TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	~~Q~~ R	1, 2, 3
c. Reactor Vessel Pressure-Low (LPCI Permissive)	NA	M	~~Q~~ R	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay <?(Minimum Flow)	NA	M	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	~~Q~~ R	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	S	M	R	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High (Pump Running)	NA	M	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	M	~~Q~~ R	1, 2, 3
c. Reactor Vessel Water Level-High, Level 8	NA	M	~~Q~~ R	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	NA	M	Q	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High (Pump Running)	NA	M	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low (Minimum Flow)	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	~~S~~ NA	M	R	1, 2, 3, 4**, 5**

No relevant indication on front of relay. at WNP-2

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~150~~ 128 psig.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required when ESF equipment is required to be OPERABLE.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per trip system requirement for one trip function in one trip system, restore the inoperable channel to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1. Each ATWS-RPT system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low, Level 2	1
2. Reactor Vessel Pressure - High	1

(a) One channel may be placed in an inoperable status for up to 2 hours for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ -{50} inches*	≥ -57 inches
2. Reactor Vessel Pressure - High	≤ {1135} psig	≤ {1150} psig

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

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TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	M	R
2. Reactor Vessel Pressure - High	(NA)	M	R

The channel calib. frequency should be extended to 18 mo (R) rather than quaterly to be consistent with other protection systems. The problem with pressure switch drift has been resolved at WNP-2 by changing manufacturers.

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END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

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

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine governor valve channel and one turbine throttle valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine governor valve channels or two turbine throttle valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3 using an MCPR limit increased by 0.07.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3 using an MCPR limit increased by 0.07.

Changes from Standard

1.) Modification of the action statements make the consequences of EOC-RPT instrumentation inoperability consistent with our plant-specific licensing basis summarized by our FSAR Chapter 15 analyses. (Action d & e)

2.) The modification to the EOC-RPT trip system response time surveillance conforms the requirement with the definition of EOC-RPT response time (table 3.3.4.2-3)

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SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

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

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine governor valve fast closure or turbine throttle valve closure, such that both types of channel inputs are tested at least once per 36 months. ~~(The time allotted for breaker arc suppression, ( ) ms, shall be verified by test at least once per 60 months.)~~

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TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)
1. Turbine Throttle Valve - Closure	2{b}
2. Turbine Governor Valve - Fast Closure	2{b}

(a) A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

(b) This function shall be automatically bypassed when turbine first stage pressure is less than or equal to ~~{275}~~ 165 psig, equivalent to THERMAL POWER less than {30}% of RATED THERMAL POWER.

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TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Throttle Valve-Closure	$\leq$ 5% closed	$\leq$ 7% closed
2. Turbine Governor Valve-Fast Closure	$\geq$ ~~980~~ psig 1250	$\geq$ ~~[illegible]~~ psig 1000

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TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milliseconds)
1. Turbine Throttle Valve-Closure	≤ ~~(100)~~ 97
2. Turbine Governor Valve-Fast Closure	≤ ~~(100)~~ 97

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TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Throttle Valve-Closure	M*	R
2. Turbine Governor Valve-Fast Closure	M*	R

*Including trip system logic testing.

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3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, 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 RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

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

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TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
a. Reactor Vessel Water Level - {Low Low, Level 2}	2	50
b. Reactor Vessel Water Level - High, Level {8}	2[{b}]	51
c. Condensate Storage Tank Water Level - Low Low	{2}[{c}]	~~52~~ 54
d. Manual Initiation	1[{d}]	53

(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 other OPERABLE channel in the same trip system is monitoring that parameter.
(b) One trip system with two-out-of-~~three~~ two logic.
(c) One trip system with one-out-of-two logic.
(d) One trip system with one channel.

per GE FCD 761E221AD Rev.4

It is not necessary to declare the RCIC system inoperable, because of the loss of the CST as the preferred source. The proposed action is consistent w/ HPCS and merely aligns the system to a safety grade source (suppression pool).

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TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition within one hour or declare the RCIC system inoperable.

b. For both trip systems, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement, declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION 53 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.

ACTION 54 - With the minimum number of operable channels one less than the total number of channels, place the inop. channel in the tripped condition in 1 hour*; operation may then continue until performance of the next required Channel Functional Test.

* The provisions of 3.0.4 are not applicable

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TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - (Low Low, Level 2)	≥ -~~38~~ 50 inches*	≥ -57 inches
b. Reactor Vessel Water Level - High, Level (8)	≤ ~~55.5~~ 54.5 inches*	≤ 56 inches
c. Condensate Storage Tank Level - Low Low **	≥ ~~( ) inches~~ 448'3"	≥ ~~( ) inches~~ 448'3"
d. Manual Initiation	NA	NA

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

** Provides automatic transfer from CST to Suppression Pool

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TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - (Low Low, Level 2)	S	M	R
b. Reactor Vessel Water Level - High, Level (8)	S	M	R
c. Condensate Storage Tank Level - Low Low	{S}	M	{R}
d. Manual Initiation	NA	~~(M)(a)~~ {R}	NA

~~(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 actuation.~~

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3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, 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 Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels 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 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[(a)]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in[(b)]	3	2	61
	2	5	61
b. Upscale[(c)]	3	2	61
	2	5	61
c. Inoperative[(c)]	3	2	61
	2	5	61
d. Downscale[(d)]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in [((e))]	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[(e)]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypass	1	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62

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TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

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

a. The RBM shall be automatically bypassed when a peripheral control rod is selected [or the reference APRM channel indicates less than [30]% of RATED THERMAL POWER].

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range [3] or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

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TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale (a)	≤ 0.66 W + ~~{~~40~~}~~% − 0.66ΔW	≤ 0.66 W + ~~{~~43~~}~~% − 0.66ΔW
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale (a)	≤ 0.66 W + ~~{~~42~~}~~% −.66ΔW*	≤ 0.66 W + ~~{~~45~~}~~% −.66ΔW*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ ~~5~~ 2 x 10^5 cps	≤ ~~5~~ 4 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 2 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ ~~(10) gallons~~ 527'1½" ele.	≤ ~~(10) gallons~~ 527'1½" ele
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	≤ 111/125 divisions of full scale
b. Inoperative	NA	NA 11
c. Comparator	≤ {10}% flow deviation	≤ ~~{11}~~% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W) and ΔW. The trip setting of this function must be maintained in accordance with Specification 3.2.2.

Footnote to Table 3.3.6.2

(a) ΔW = Difference in indicated drive flow (in percent of rated drive flow) between two-loop and single-loop operation at the same core flow

ΔW = 0 for two recirculation loop operation

= (Later) for single-loop operation

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TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[(b)(c)], M[(c)]	Q	1*
b. Inoperative	NA	S/U[(b)(c)], M[(c)]	NA	1*
c. Downscale	NA	S/U[(b)(c)], M[(c)]	Q	1*
2. APRM				
a. Flow Biased Neutron Flux Upscale	NA	S/U[(b)], M	(Q)	1
b. Inoperative	NA	S/U[(b)], M	NA	1, 2, 5
c. Downscale	NA	S/U[(b)], M	(Q)	1
d. Neutron Flux - Upscale, Startup	NA	S/U[(b)], M	(Q)	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)], W	NA	2, 5
b. Upscale	NA	S/U[(b)], W	Q	2, 5
c. Inoperative	NA	S/U[(b)], W	NA	2, 5
d. Downscale	NA	S/U[(b)], W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)], W	NA	2, 5
b. Upscale	NA	S/U[(b)], W	Q	2, 5
c. Inoperative	NA	S/U[(b)], W	NA	2, 5
d. Downscale	NA	S/U[(b)], W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q (Float Switch) H (Transmitter)	R	1, 2, 5**
b. Scram Trip Bypass	NA	M	NA	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[(b)], M	Q	1
b. Inoperative	NA	S/U[(b)], M	NA	1
c. Comparator	NA	S/U[(b)], M	Q	1

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TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

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INSTRUMENTATION

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

NOTE: Explaination of changes to this section on back.

This specification submitted with environmental tech. spec's and provided herein for info and completeness.

Table 3.3.7.1-1 "RADIATION MONITORING INSTRUMENTATION" was modified to be consistant with other nuclear plant technical specifications. The following changes were made:

(1) Off-Gas Pre-treatment Radiation Monitor - located on table 3.3.7.12-1 & table 4.3.7.12-1.
(2) Reactor Build vent Radiation Monitor: - located on table 3.3.7.12-1 and table 4.3.7.12-1 "REACTOR BUILDING ELEVATED RELEASE MONITOR".
(3) Off-Gas Post-Treatment Radiation Monitor - located on table 3.3.7.12-1 and table 4.3.7.12-1.
(4) Standby Gas Treatment System Exhaust Monitor - This system has the "REACTOR BUILDING ELEVATED RELEASE MONITOR" as it's monitor - located on tables 3.3.7.12-1 & 4.3.7.12-1.
(5) Area Monitors (b). Control Room Direct Radiation Monitor - This is currently not a techspec item consistant with similar nuclear power stations.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Ventilation Radiation Monitor	2/(intake)	1,2,3,5 and *	≤ ~~(4) mR/hr~~ 5000 cpm	70
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	1	#	≤ (15) mR/hr[(a)]	71
2) Spent Fuel Storage Pool	1	##	≤ (15) mR/hr[(a)]	71

TABLE 3.3.7.1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATION

*When the main condenser air evacuation system is in operation.

(a) Alarm only.

#With fuel in the new fuel storage vault.
##With fuel in the spent fuel storage pool.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 70 -

a. With one of the required monitors inoperable, place the inoperable channel in the (downscale) tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the (isolation) mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the (isolation) mode of operation within one hour.

ACTION 71 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Main Control Room Ventilation Radiation Monitor	S	M	R	1, 2, 3, 5 and *
7. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	#
2) Spent Fuel Storage Pool	S	M	R	##

TABLE 4.3.7.[illegible] (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

#With fuel in the new fuel storage vault.
##With fuel in the spent fuel storage pool.
*When the main condenser air evacuation system is in operation.

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.2 The seismic monitoring instrumentation shown in Table 3.3.7.2-1~~(*)~~ shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.2.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.2-1.

4.3.7.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to (0.01) g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. In lieu of any other report required by Specification 6.9.1, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety.

~~(*This instrumentation may be shared with additional units at a common site provided seismic instrumentation and corresponding technical specifications meet the recommendations of Regulatory Guide 1.12, April 1974.)~~ N/A

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TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. Reactor Building foundation (Ele. 422)	±1g	1
b. Containment Drywell floor (Ele. 501')	±1g	1
c. Free field	±1g	1
2. Triaxial Peak Accelerographs		
a. Reactor vessel head	±5g	1
b. HPCS injection piping	±5g	1
c. Standby service water pump house	±5g	1
3. Triaxial Seismic Switches		
a. Reactor Building foundation	.025 to .25g	1[(a)]
4. Triaxial Response-Spectrum Recorders		
a. Reactor Building ~~floor~~ foundation	1.6 to 34	1[(a)]
b. HPCS injection line piping support	1.6 to 34	1
c. Reactor Building ~~foundation~~ Refuel Floor	1.6 to 34	1
d. Radwaste Building foundation	1.6 to 34	1

(a) With reactor control room ~~indication and~~ annunciation.

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TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs			
a. Reactor Building foundation	M	SA	R
b. Containment Drywell floor	M	SA	R
c. Free field	M	SA	R
2. Triaxial Peak Accelerographs			
a. Reactor vessel head	NA	NA	R
b. HPCS injection piping	NA	NA	R
c. Standby service water pump house	NA	NA	R
3. Triaxial Seismic Switches			
a. Reactor Building foundation	M[(a)]	SA	R
4. Triaxial Response-Spectrum Recorders			
a. Reactor Building ~~floor~~ foundation	M	SA	R
b. HPCS injection line piping support	NA	~~SA~~ NA	R
c. Reactor Building ~~foundation~~ Refuel floor	N/A	~~SA~~ NA	~~R~~
d. Radwaste Building foundation	N/A	~~SA~~ NA	~~R~~

(a) Except seismic trigger.

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INSTRUMENTATION

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.3 The meteorological monitoring instrumentation channels ~~shown in Table~~ 3.3.7.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more meteorological monitoring instrumentation channels inoperable for more than 7 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.3 Each of the above required meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.3-1.

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
a. Wind Speed	
1. Elev. 33 ft.	1
2. Elev. 245 ft.	1
b. Wind Direction	
1. Elev. 33 ft.	1
2. Elev. 245 ft.	1
c. Air Temperature Difference	
1. Elev. 33/245 ft.	1

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TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 33 ft.	D	SA
2. Elev. 245 ft.	D	SA
b. Wind Direction		
1. Elev. 33 ft.	D	SA
2. Elev. 245 ft.	D	SA
c. Air Temperature Difference		
1. Elev. 33/245 ft.	D	SA

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INSTRUMENTATION

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown monitoring instrumentation channels shown in Table 3.3.7.4-1 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

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TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	C61-P001	1
2. Reactor Vessel Water Level	C61-P001	1
3. Safety/Relief Valve ~~Position~~ TEMPERATURE, ~~8~~ 2 valves	~~C61-P001~~ H22-P100	1/valve
4. Suppression Chamber Water Level	H22-P100	1
5. Suppression Chamber Water Temperature (4)	H22-P100	1
6. Service Water Pump B Discharge Pressure	H22-P100	1
7. Drywell Pressure, Low Range	H22-P100	1
8 Drywell Pressure, High Range	H22-P100	1
9. ~~8~~. Upper Drywell Temperature	H22-P100	1
10. ~~9~~. RHR System B Flow	H22-P100	1
11. ~~10~~. Spray Pond B Level	H22-P100	1
12. ~~11~~. Spray Pond B Temperature	H22-P100	1
13. ~~12~~. RCIC System Flow	C61-P001	1
14. ~~13~~. RCIC Turbine Speed	C61-P001	1

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TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Safety/Relief Valve ~~Position~~ Temperature	M	~~NA~~ R
4. Suppression Chamber Water Level	M	R
5. Suppression Chamber Water Temperature (4)	M	R
6. Service Water Pump B Discharge Pressure	M	R
7. Drywell Pressure, Low Range	M	R
8. Drywell Pressure, High Range	M	R
9. ~~8.~~ Upper Drywell Temperature	M	R
10. ~~9.~~ RHR System Flow	M	R
11. ~~10.~~ Spray Pond B Level	M	R
12. ~~11.~~ Spray Pond B Temperature	M	R
13. ~~12.~~ RCIC System Flow	M	R
14. ~~13.~~ RCIC Turbine Speed	M	R

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LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

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TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	80
2. Reactor Vessel Water Level	{2}	1	1, 2	80
3. Suppression Chamber Water Level	2	1	1, 2	80
4. Suppression Chamber Water Temperature	2/~~sector~~	1/~~sector~~	1, 2	80
5. Suppression Chamber Air Temperature	2	1	1, 2	80
6. Drywell Pressure	{2}	1	1, 2	80
7. Drywell Air Temperature	2	1	1, 2	80
8. Drywell Oxygen Concentration	2	1	1, 2	80
9. Drywell Hydrogen Concentration ~~Analyzer and Monitor~~	2	1	1, 2	80
10. Safety/Relief Valve Position Indicators	2/valve	1/valve	1, 2	80
~~(11. In Core Thermocouples~~ N/A for WNP-2	~~(4)/(1 per core quadrant)~~	~~(2)/(1 each of 2 core quadrants)~~	~~1, 2~~	~~80)~~
11. Suppression Chamber Pressure	2	1	1, 2	80
12. Condensate Storage Tank Level	2*	1	1, 2	80
13. Main Steam Line Isolation Valve Leakage Control System Pressure	2	1	1, 2	80
~~14. Drywell Spray Flow~~ same as RHR Flow at WNP-2	~~2~~	~~1~~	~~1, 2~~	~~80~~

change numbers

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TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
15. Neutron Flux:				
APRM	2	1	1, 2	80
~~IRM~~	~~2~~	~~1~~	~~1, 2~~	~~80~~
SRM	2	1	1, 2	80
16. RCIC Flow	1	*1	1, 2	80
17. HPCS Flow	1	*1	1, 2	80
~~18. LPCI Flow~~ Sameas RHR Flow	~~1~~	~~*~~	~~1, 2~~	~~80~~
19. LPCS Flow	1	*1	1, 2	80
20. Standby Liquid Control System Flow	X1	1	1, 2	80
21. Standby Liquid Control System Tank Level	X1	1	1, 2	80
22. RHR Flow	1/loop	*1	1, 2	80
23. RHR Heat Exchanger Outlet Temperature	1/heat exchanger	1	1, 2	80
24. ~~ESF Cooling~~ Standby Service Water Flow	1/loop	1	1, 2	80
25. ~~ESF Systems Cooling Water Temperature~~ Standby Service Water Spray Pond Temperature	~~1/loop~~ 2	1	1, 2	80
26. Radioactive Liquid Tank Level	2/tank	1	1, 2	80
27. Emergency Ventilation Damper Position	2/~~damper~~duct	1/duct	1, 2	80
28. Standby Power and Other Energy Sources	2/source	1	1, 2	80
29. Airborne Particulate and Halogen Release	2	1	1, 2	80
~~30. Control Rod Position~~ delete, Non-class 1E & covered in 3/4.1.3.7 adequately.	~~1/rod~~	~~1/group~~	~~1, 2~~	~~80~~
31. Primary Containment Valve Position	2/~~valve~~line	1/line	1, 2	80
32. Primary Containment Gross Radiation Monitors	2/range	1/range	1, 2, 3	81
33. Primary Coolant Radiation Monitor ~~*~~	{1}	1	1, 2, 3	81
34. Effluent Noble Gas Radiation Monitor #	{1}	1	1, 2, 3	81

#High range noble gas monitor~~s~~

* A single CST is sufficient as provided for in specification 3/4.5.3

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Table 3.3.7.5-1 (Continued)
ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1, 2
2. Reactor Vessel Water Level	M	R	1, 2
3. Suppression Chamber Water Level	M	R	1, 2
4. Suppression Chamber Water Temperature	M	R	1, 2
5. Suppression Chamber Air Temperature	M	R	1, 2
6. Primary Containment Pressure	M	R	1, 2
7. Drywell Air Temperature	M	R	1, 2
8. Drywell Oxygen Concentration	M	R	1, 2
9. Drywell Hydrogen Concentration ~~Analyzer and Monitor~~	M	Q*	1, 2
10. Safety/Relief Valve Position Indicators	M	R	1, 2
~~11. In-Core Thermocouples~~ N/A for WNP-2	~~M~~	~~R~~	~~1, 2~~
11. Suppression Chamber Pressure	M	R	1, 2
12. Condensate Storage Tank Level	M	R	1, 2
13. Main Steam Line Isolation Valve Leakage Control System Pressure	M	R	1, 2
~~14. Drywell Spray Flow~~	M	R	1, 2
15. Neutron Flux:			
APRM	M	R	1, 2
~~IRM~~	~~M~~	~~R~~	~~1, 2~~
SRM	M	R	1, 2
16. RCIC Flow	M	R	1, 2
17. HPCS Flow	M	R	1, 2
~~18. LPCI Flow~~	M	R	1, 2
19. LPCS Flow	M	R	1, 2

change numbers

TABLE 4.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
20. Standby Liquid Control System Flow	M	R	1, 2
21. Standby Liquid Control System Tank Level	M	R	1, 2
22. RHR Flow	M	R	1, 2
23. RHR Heat Exchanger Outlet Temperature	M	R	1, 2
24. ~~ESF Systems Cooling~~ Standby Service Water Flow	M	R	1, 2
25. ~~ESF Systems Cooling Water~~ Standby Service Water Spray Pond Temperature	M	R	1, 2
26. Radioactive Liquid Tank Level	M	R	1, 2
27. Emergency Ventilation Damper Position	M	R	1, 2
28. Standby Power and Other Energy Sources	M	R	1, 2
29. Airborne Particulate and Halogen Release	M	R	1, 2
~~30. Control Rod Position~~ functional requirements adequately covered in 3/4.1.3.7.	~~M~~	~~R~~	~~1, 2~~
31. Primary Containment Valve Position	M	R	1, 2
32. Primary Containment Gross Radiation Monitors	M	R**	1, 2, 3
33. Primary Coolant Radiation Monitor *	M	R	1, 2, 3
34. Effluent Noble Gas Radiation Monitor #	M	R	1, 2, 3

*Using sample gas containing:
a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

**CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

#High range noble gas monitors.

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK at least once per:

a) 12 hours in CONDITION 2*, and

b) 24 hours in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least ~~3~~ 0.5 cps with the detector fully inserted.

see justification for 3/4.9.2

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

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INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe system shall be OPERABLE with:

a. Three movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all three detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or (TPF) (MFLPD).

ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the above applicable monitoring or calibration functions.

*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

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INSTRUMENTATION

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 *Two independent chlorine detection system subsystems shall be OPERABLE with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm within ~~5 seconds~~ 10 seconds.

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

a. With one chlorine detection subsystem inoperable, restore the inoperable detection subsystem to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system train in the isolation mode of operation.

b. With both chlorine detection subsystems inoperable, within one hour initiate and maintain operation of at least one control room emergency filtration system train in the isolation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each of the above required chlorine detection system subsystems shall be demonstrated OPERABLE by performance of

a. CHANNEL CHECK at least once per 12 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

* The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4 or 5.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.9 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.9-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With the number of OPERABLE fire detection instruments in one or more zones:

1. Less than, but more than one-half of, the Total Number of Instruments shown in Table 3.3.7.9-1 for Function A, restore the inoperable Function A instrument(s) to OPERABLE status within 14 days or within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, ~~unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.7).~~

2. One less than the Total Number of Instruments shown in Table 3.3.7.9-1 for Function B, or one-half or less of the Total Number of Instruments shown in Table 3.3.7.9-1 for Function A, or with any two or more adjacent instruments inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, ~~unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.7).~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

WNP-2 has not fire protection devices inside containment since the atmosphere is inerted to less than 3.5% O_2.

SURVEILLANCE REQUIREMENTS

4.3.7.9.1 Each of the above required fire detection instruments which are accessible during unit operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during unit operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.7.9.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.7.9.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

Time change from 5 to 10 seconds represents a plant specific design parameter and is consistent with the WNP-2 FSAR and the manufacturer of the Cl_2 detectors stated response time.

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION

	ID	SD	TD	UD
REACTOR BUILDING 422'-3"				
CRD PUMP ROOM	4X			
AUX COND. PUMPROOM	2X			
REACTOR BUILDING 441'-0"				
RAIL ROAD AIR LOCK		4X		
REACTOR BUILDING 444'-0"				
RHR PUMP ROOM R2	3X			
RHR PUMP ROOM R1	3X			
RHR PUMP ROOM R4	3X			
RCIC PUMP ROOM R3	3X			
LPCS PUMP ROOM R5	2X			
HPCS PUMP ROOM R6	3X			
REACTOR BUILDING 471'-0"				
MCC ROOM	1X			
GENERAL AREA	24X			
REACTOR BUILDING 501'-0"				
GENERAL AREA	23X			
REACTOR BUILDING 522'-0"				
MCC ROOM DIV. II	1X			
GENERAL AREA	28X			

TABLE 3.3.7.9-1 (CONT)

INSTRUMENT LOCATION	ID	SD	TD	U
REACTOR BUILDING 548'-0"				
FUEL Pool HT. EXCHGR RoomA & Pump Room	1X			
GENERAL AREA	30X			
REACTOR BUILDING 572'-0"				
Hydrogen RECOMBINER CONT. RM DIV II	2X			
RHR HT. EXCHGR RM 1A	1X			
RHR HT. EXCHGR RM 1B	1X			
GENERAL FLOOR AREA	25X			
REACTOR BUILDING 606'-10.5"				
GENERAL FLOOR AREA				6.
RADWASTE & CONTROL BUILDING 467'-0"				
ELECT. EQUIP. RM. NO. 1	2X			
BATTERY RM. NO. 1	4X			
SWITCHGEAR RM. NO. 1	3X			
ELECT. EQUIP. RM. NO. 2	3X			
BATTERY RM. NO. 2	2X			
SWITCH GEAR RM. NO. 2	3X			
REMOTE SHUTDOWN RM.	1X			
CORRIDOR C-205	5X			
RADWASTE & CONTROL BUILDING 484'-0"				
CABLE SPREADING RM.	36Y			

TABLE 3.3.7.9-1 (CONT)

INSTRUMENT LOCATION

	ID	SD	TD	UD
RADWASTE & CONTROL BUILDING 501'-0"				
CABLE CHASE	5Y			
CONTROL ROOM (CEILING)	12X	1X		
RADWASTE & CONTROL BUILDING 525'-0"				
CABLE CHASE	6Y			
UNIT A - AIR CONDITIONING RM	5X			
UNIT B - AIR CONDITIONING RM	5X			
TURBINE GENERATOR CORRIDOR 441'-0"				
TG-1 CORRIDOR	18Y/6X			
DIESEL GENERATOR BUILDING 441'-0"				
1A DIESEL GENERATOR ROOM	2X	4X	4Y	
1A DIESEL DAY TANK ROOM			1X/1Y	
1A DIESEL OIL TANK PUMP ROOM			1X/1Y	
1B DIESEL GENERATOR ROOM	2X	4X	4Y	
1B DIESEL DAYTANK ROOM			1X/1Y	
1B DIESEL OIL TANK PUMP ROOM			1X/1Y	
HPCS DIESEL GENERATOR ROOM	2X	4X	4Y	
HPCS DIESEL ~~[illegible]~~ DAY TANK ROOM			1X/1Y	
HPCS DIESEL OIL TANK PUMP ROOM			1X/1Y	

TABLE 3.3.7.9-1 (CONT.)

INSTRUMENT LOCATION

	ID	SD	TD	UD
STANDBY SERVICE WATER PUMP HOUSE 1A				
PUMP HOUSE			1X	
ELECT. VAULT	1X			
STANDBY SERVICE WATER PUMP HOUSE 1B				
PUMP HOUSE			1X	
ELECT VAULT	1X			

ID - IONIZATION DETECTOR
SD - PHOTOELECTRIC DETECTOR (Smoke)
TD - Thermal DETECTOR
UD - Ultraviolet (Flame) DETECTOR
SMD - DUCT IONIZATION DETECTOR

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LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING FOR OPERATION

3.3.7.11 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.7.11-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters described in the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: At all times.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.11-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Semiannual Radioactive Effluent Release Report.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.11 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.11-2.

This spec. provided to support related changes.

TABLE 3.3.7.11-1

Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Channels Operable	Action
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1	100
2. Gross Radioactivity Monitors Not Providing Automatic Termination of Release		
a. Service Water System Effluent Line	1	101
b. RHR Service Water System Effluent Line	1/Loop	101
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1	102
b. Plant Discharge - Blowdown Line	1	102

TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

ACTION 100 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 101 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 102 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases.

TABLE 4.3.7.11-2

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Radioactivity Monitors Providing Automatic Termination of Release				
a. Liquid Radwaste Effluent Line	P*	P	R(3)	Q(1)
2. Gross Radioactivity Monitors Not Providing Automatic Termination of Release				
a. Service Water System Effluent Line	D	M	R(3)	Q(2)
b. RHR Service Water System Effluent Line	D	M	R(3)	Q(2)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Plant Discharge-Blowdown Line	D(4)	N.A.	R	Q

*Perform CHANNEL CHECK at least one per 24 hours if discharge valve interlocks referenced in Table 3.3.7.10-1 are not functioning.

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the follwing conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement.)

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters described in the ODCM.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Semiannual Radioactive Effluent Release Report.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.12-2.

This spec. provided to support related changes.

TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicable	Action
1. Main Condenser Off-Gas Post-Treatment Monitor			
a. Gross Gamma Detector Alarm and Automatic Isolation of the Off-Gas System Outlet and Drain Valves.	2	**	110
2. Main Condenser Offgas Pre-Treatment Monitor			
a. Gamma sensitive ion-chamber located upstream of holdup line	1	**	115
3. Reactor Building Elevated Release Monitor			
a. Noble Gas Activity Monitor	1	*	110
b. Iodine Sampler	1	*	112
c. Particulate Sampler	1	*	112
d. Effluent System Flow Rate Monitor	1	*	114
e. Sampler Flow Rate Monitor	1	*	114
4. Turbine Building Ventilation Exhaust Monitor			
a. Noble Gas Activity Monitor	1	*	110
b. Iodine Sampler	1	*	112
c. Particulate Sampler	1	*	112
d. Effluent System Flow Rate Monitor	1	*	114
e. Sampler Flow Rate Monitor	1	*	114

TABLE 3.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicable	Action
5. Rad-Waste Building Ventilation Exhaust			
a. Noble Gas Activity Monitor	1	*	110
b. Iodine Sampler	1	*	112
c. Particulate Sampler	1	*	112
d. Effluent System Flow Rate Monitor	1	*	114
e. Sampler Flow Rate Monitor	1	*	114
6. Main Condenser Off-Gas Treatment Explosive Gas Monitoring			
a. Hydrogen	2	**	111

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATION

* At all times
** During Main Condenser Off-Gas Treatment System Operation

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for noble gas gamma emitters within 24 hours.

ACCTION 111 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least one per 4 hours and analyzed within the following 4 hours. If the recombiner(s) temperature remains constant and THERMAL POWER has not changed, the grab sample collection frequency may be changed to 8 hours.

ACTION 112 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11.2-1.

ACTION 113 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 114 - With the number of channels OPERABLE less than required by the minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 115 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases to the environment may continue for up to 72 hours provided:

a. The offgas system is not bypassed, and
b. The Turbine Building vent noble gas activity monitor is OPERABLE;

Otherwise, be in at least STARTUP/HOT STANDBY within 12 hours.

TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Main Condenser Off-Gas Post Treatment Monitor					
a. Gross gamma detector alarm and automatic isolation of the off gas system outlet and drain valves	D	D	Q(1)	R(2)	*
2. Main Condenser Off-Gas Pre-Treatment Monitor					
a. Bamma sensitive ion-chamber located upstream of hold up line	D	M	Q(1)	R(2)	**
3. Reactor Building Elevated Release Monitor					
a. Noble Gas Activity Monitor	D	M	Q(1)	R(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	Q	R	*
e. Sampler Flow Rate Monitor	D	N.A.	Q	R	*
4. Turbine Building Ventilation Exhaust Monitor					
a. Noble Gas Activity Monitor	D	M	Q(1)	R(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	Q	R	*
e. Sampler Flow Rate Monitor	D	N.A.	Q	R	*

TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
5. Rad-Waste Building Ventilation Exhaust Monitor					
a. Noble Gas Activity Monitor	D	M	Q(1)	R(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	Q	R	*
e. Sampler Flow Rate Monitor	D	N.A.	Q	R	*
6. Main Condenser Off-Gas Treatment System Explosive Gas Monitoring System					
a. Hydrogen Monitor	D	N.A.	M	Q(3)	**

TABLE NOTATION

* At all times
** During Main Condenser Off-Gas Treatment System Operation

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.

2. Circuit failure.

3. Instrument controls not set in the Operate mode.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by ther National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. Operating plants may substitute previously established calibration procedures for this requirement.

(3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. 0.5 volume percent hydrogen, balance nitrogen, and

2. Four volume percent hydrogen, balance nitrogen.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.8 Both the mechanical and electrical turbine overspeed protection systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one turbine governor valve or one turbine throttle valve per steam chest inoperable and not closed, restore the inoperable valve to OPERABLE status within 72 hours, isolate the affected steam chest from the steam supply, or isolate the turbine from the steam supply within the next 6 hours.

b. With one turbine interceptor valve or one turbine reheat stop valve inoperable, restore the inoperable valve to OPERABLE status within 72 hours, or close at least one valve in the affected steam line or isolate the turbine from the steam supply within the next 6 hours.

c. With either of the overspeed protection systems otherwise inoperable, restore the inoperable system to OPERABLE status within 72 hours or isolate the turbine from the steam supply within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

4.3.8.2 The above required turbine overspeed protection systems shall be demonstrated OPERABLE:

a. At least once per 7 days by:

Cycling each of the following valves through at least one complete cycle from the running position:

1. Four high pressure turbine throttle valves;
2. Six low pressure turbine reheat stop valves;
3. Four high pressure turbine governor valves; and
4. Six low pressure turbine interceptor valves.

SURVEILLANCE REQUIREMENTS (Continued)

b. Once per 18 months or once per six month period during a reactor startup, whichever is shorter, perform a Logic System Functional Test for the electrical overspeed protection controller system.

c. By monthly verifying OPERABILITY of the mechanical overspeed trip mechanism.

d. Once per 18 months or once per six month period during a reactor startup, whichever is shorter, perform a Logic System Functional Test of the mechanical overspeed trip system.

e. At least once per 31 days by direct observation of the movement of each of the above valves through at least one complete cycle from the running position.

f. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems, and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws on excessive corrosion are found, all other valves of that type shall be inspected.

This LCO has been rewritten to reflect the design of the Westinghouse supplied Turbine/Generator overspeed protection systems. The change incorporates the requirements and content of the previously issued LCO and the vendor recommended tests and frequencies.

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3/4.3.9 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM

~~LIMITING~~ CONDITION FOR OPERATION

3.3.9 The feedwater system/main turbine trip system actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a feedwater system/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. For the feedwater system/main turbine trip system:

X With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

delete (cover)

2. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

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SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater system/main turbine trip system 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 4.3.9.1-1.

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

Justification:

Feedwater turbine / Main Turbine Trip System consists of 3 channels with 2 out of 3 logic. Therefore there is no need for action b2 since the turbines should already be tripped.

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TABLE 3.3.9-1

FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS
1. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level (8)	3	1

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TABLE 3.3.9-2

FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level 8	≤ 54.5 inches*	≤ 56.0 inches

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

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TABLE 4.3.9.1-1

FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level (8)	NA	M	R	1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, insert attached ~~immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours.~~

b. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure (at the hydraulic control unit), and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to ~~10%~~ 11% of full stroke per second opening, and

2. Less than or equal to ~~10%~~ 11% of full stroke per second closing.

Insert attached 4.4.1.2

*See Special Test Exception 3.10.4.

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TECHNICAL SPECIFICATION 3/4 4.1, PAGE 3/4 4-1

3.4.1.1 ACTION a

Delete "...immediately initiate... within the next 12 hours."

Insert following "...in operation," :

operation may continue provided adjustments are made for:

1. MCPR Fuel Cladding Safety and operating limits (Specifications 2.1.2 and 3/4.2.3)
2. APRM scram and rod block setpoints (Table 2.2.1-1 and Specification 3/4 2.2). RBM setpoints (Table 3.3.6-2)
3. MAPLHGR (Specification 3/4 2.1)
4. Immediate reduction of operating recirculation loop ~~pump speed to less than (later)% of rated or~~ flow control valve position to less than (later)% of rated.

4.4.1.2 With one recirculation loop inoperable, verify that recirculation ~~pump speed and/or~~ flow control valve position is less than that specified in Specification 3.4.1.1.6 at least once per 24 hours.

The provisions of Specification 3.0.4 are not applicable.

Changes to Standard

1) Indicated modifications are a result of the plant specific analyses to justify single loop operation.

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by measuring and recording each of the below specified parameters and verifying that no two of the following conditions occur when the recirculation loops are operating at the same flow control valve position.

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from either the:

   1. Established THERMAL POWER-core flow relationship, or

   2. Established core plate differential pressure-core flow relationship.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

Washington Nuclear Unit 2

REACTOR COOLANT SYSTEM

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RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

For two (2) recirculation loop operation

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to ~~70%~~ 80 of rated core flow.

b. 10% of rated recirculation flow with core flow less than ~~70%~~ 80 of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare the recirculation loop with the lower flow not in operation and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

Revise to use La Salle's Tech. Spec's

The only substantive difference is in item 4.4.1.2. b where La Salle has expanded the meaning of "recirculation loop flow measurements", to be more specific by stating the 2 accepted methods of alternate core flow indication. This is easily justified since it ~~is~~ makes the section more specific and less obscure.

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REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F, and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

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

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

Changes to Standard

1) Indicated changes are a result of plant-specific analyses to justify single reactor loop operation and setpoints/design specifications.

G.E. DOCUMENT 22A4106 REV (MPL B22-5030 DC) "REACTOR VESSEL OVERPRESSURE PROTECTION SYSTEM" SPECIFIES, THE NUMBER OF SAFETY VALVES REQUIRED TO FUNCTION FOR THE WORST CASE TRANSIENTS TO BE 12 FOR WNP-2.

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REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

~~LIMITING CONDITION~~ FOR OPERATION

3.4.2.~~1~~ The safety valve function of at least ~~(16)~~ 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1148 psig ±1%
4 safety/relief valves @ 1175 psig ±1%
4 safety/relief valves @ 1185 psig ±1%
4 safety/relief valves @ 1195 psig ±1%
4 safety/relief valves @ 1205 psig ±1%

+0%
-2%

THE CORRECT TOLERAN IS GIVEN BY G.E. LICENSING ACTION NOTE #31 OF JAN 7, 1982. (Over for Justification)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 90°F, close the stuck open safety/relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is ~~(105)~~°F or greater, place the reactor mode switch in the Shutdown position. 110°F

110°F PROVIDES CONSISTEN WITH SPECIFICATION 3.6.2. SUPPRESSION CHAMBER

c. With one or more safety/relief valve acoustic monitors} inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE ~~with the setpoint verified to be ( )~~ by performance of a:

a. CHANNEL ~~(FUNCTIONAL TEST)~~ {CHECK} at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months {**}

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

{**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.}

SAFETY
RELIEF VALVE TOLERANCE

THE CORRECT TOLERANCE IS +0 TO -2% AS GIVEN BY THE G.E. LICENSING ACTION NOTICE #31 OF JAN 7, 1982. THE CROSBY SRVs NEED A TOLERANCE OF +0 TO -2% IN ORDER TO STAY BELOW THE +1% DRIFT LIMIT USED IN TRANSIENT ANALYSES. THE ASME SECTION III CODE STATES THAT THE SRVs SHALL HAVE A SETPOINT TOLERANCE OF ±1% OR GREATER IF EVALUATED IN THE OVERPRESSURE PROTECTION REPORT (WHICH OURS ARE) AND IDENTIFIED IN THE DESIGN SPECIFICATION (OUR DESIGN SPEC, GE DOCUMENT 22A6441, PAGE 6, SPECIFIES A TOLERANCE OF +0 TO -2%).

4.4.21 THE SETPOINT WILL BE SUPPLIED ~~BY~~ IN THE PROCEDURES WHICH WILL BE USED TO PERFORM THE CHANNEL CHECK AND THE CHANNEL CALIBRATION

Low-Low Set Function is not applicable to the WNP-2 containment design. Ref: WNP-2 FSAR Volume entitled "Plant Design Assessment Report for SRV and LOCA Loads"

~~SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:~~

Valve No.	Low-Low Set Function Setpoint* (psig) ± 1% Open	Low-Low Set Function Setpoint* (psig) ± 1% Close	Relief Function Setpoint* (psig) ± 1% Open	Relief Function Setpoint* (psig) ± 1% Close
______	(1033)	(926)	______	______
______	(1073)	(936)	______	______
______	(1113)	(946)	______	______
______	(1113)	(946)	______	______
______	(1113)	(946)	______	______

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

~~ACTION:~~

a. ~~With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

b. ~~With the relief valve function and/or the low-low set function of more than one of the above required reactor coolant system/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:~~

a. ~~CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 days.~~

b. ~~CHANNEL CALIBRRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.~~

~~*The life setting pressure shall correspond to ambient conditions of the values at nominal operating temperatures and pressures.~~

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3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system,

b. The primary containment sump flow monitoring system, and

c. The primary containment atmosphere particulate radioactivity monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere particulate and ~~particulate and~~ gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

Two detectors are utilized in the containment leak monitor. One is part of the moving filter detector which continuously monitors the sample stream for suspended particulate. The other is part of the gaseous monitor.

THIS PAGE OPEN PENDING RECEIPT OF ... FROM THE APPLICANT

REACTOR COOLANT SYSTEM

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 1 gpm leakage at a ~~reactor coolant system~~ test pressure of 950 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in unidentified leakage within any 4 hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed (manual or deactivated automatic) (or check*) valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. (over)

(*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.)

c. With any reactor coolant system pressure isolation valve (Table 3.4.3.2-1) leakage greater than the above limit, isolate the high pressure portion of the affected system or subsystem from the low pressure portion within 4 hours, declare the affected system or subsystem inoperable and take appropriate action per specification 3.5.1 or 3.7.3.

Action c. redraft Justification:

- Some systems/subsystems do not have two isolation valves accessible during operation. In most cases, the only acceptable alternate isolation valve is a manual valve located inside the "inerted" drywell.
- Low Pressure piping in the systems affected at WNP-2 are protected by relief valves, rated at 10 times greater than the specified 1 gpm limit and discharge to the suppression pool.
- Also, pressure instrumentation is available (Table 3.4.3.2-2) to warn of potential high pressure system leakage and is subject to routine surveillance.

The intent of the specification is retained and the design of the WNP-2 systems is sufficient justification for the change.

Add e. per GE

e. With any reactor coolant system unidentified leakage increase greater than 2 gpm within any 4 hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours, or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric particulate and gaseous radioactivity at least once per 12 hours,

b. Monitoring the primary containment sump flow rate or the caseous particulate radioactivity at least once per 12 hours,

c. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

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TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
HPCS-V-4	HPCS
HPCS-V-5	HPCS
LPCS-V-5	LPCS
LPCS-V-6	LPCS
RCIC-V-66	RCIC
RCIC-V-13	RCIC
RHR-V-8	RHR
RHR-V-9 /209	RHR
RHR-V-23	RHR
RHR-V-41A, B, C	RHR
RHR-V-42A, B, C	RHR
RHR-V-50A/123A, 50B/123B	RHR
RHR-V-53A, B	RHR

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES
LEAKAGE PRESSURE MONITORS

Instrument ~~VALVE~~ NUMBER	Function ~~SYSTEM~~	ALARM SETPOINT (psig)
HPCS-PS-3 (E22-N003)	Pump Suction Pressure High	80
LPCS-PS-5 (E21-N005)	Pump Discharge Pressure High	442*
RCIC-PS-21 (E51-N021)	Pump suction pressure High	60
RHR-PIS-22A,B&C (E12-N022)	Pump Discharge Pressure to RPV High	379*
RHR-PS-18 (E12-N018)	shutdown cooling suction pressure High	171*

* Water column head compensated

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.4 are not applicable.

2. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

1. With the:

   a) Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours, or

   b) Chloride concentration exceeding the limit specified in Table 3.4.4-1, restore the chloride concentration to within the limit within 24 hours, or

   perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 3.

2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 The reactor coolant shall be determined to be within the specified chemistry limit by:

a. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 hours.

b. Analyzing a sample of the reactor coolant for:

1. Chlorides at least once per:

a) 72 hours, and

b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

2. Conductivity at least once per 72 hours.

3. pH at least once per:

a) 72 hours, and

b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

c. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, for up to 31 days, obtaining an in-line conductivity measurement at least once per:

1. 4 hours in OPERATIONAL CONDITIONS 1, 2 and 3, and

2. 24 hours at all other times.

d. Performance of a CHANNEL CHECK of the continuous conductivity monitor with an in-line flow cell at least once per:

1. 7 days, and

2. 24 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

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TABLE 3.4.4-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL CONDITION	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)	PH
1	≤ 0.2 ppm	≤ 1.0	5.6 ≤ pH ≤ 8.6
2 and 3	≤ 0.1 ppm	≤ 2.0	5.6 ≤ pH ≤ 8.6
At all other times	≤ 0.5 ppm	≤ 10.0	5.3 ≤ pH ≤ 8.6

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3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

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

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

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or for more than 800 hours cumulative operating time in a consecutive 12-month period, or greater than 4.0 microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. Greater than $100/\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

b. In OPERABLE CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

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REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour*, or
2. The off-gas level, at the SJAE, increased by more than ~~10,000~~ 15,000 microcuries per second in one hour during steady state operation at release rates less than ~~75,000~~ 100,000 microcuries per second (30 min delay mixture), or
3. The off-gas level, at the SJAE, increased ~~by more than~~ 15% in one hour during steady state operation at release rates greater than ~~75,000~~ 100,000 microcuries per second (30 minute delay)

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.2 at least once per 92 days containing the results of the specific activity analysis together with the below additional information for each occurrence.

Additional Information

1. Reactor power history starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.
2. Fuel burnup by core region.
3. Clean-up flow history starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.
4. Off-gas level starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

*Not applicable during the startup test program.

Justifications:

c.2) a. The 100,000 μci/sec (30 minute delay mixture) is chosen vrs 70,000 μci/sec to give some design basis connection to this number (reference GE NEDO -10871, March 1973, Technical Derivation of 1971 BWR Design Basis Radioactive Material Source Terms).

b. The 15,000 μci/sec per hour change was selected to provide continuity to c.3 from c.2 as c.3 begins at 15% of 100,000 or 15,000. Both the 15,000 and 100,000 μci/sec values are conservative with regard to the 10CFR 100 criteria.

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TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the primary coolant system is restored to within its limits.

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curves A and A' for hydrostatic or leak testing; (2) curves B and B' for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C and C' for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20 °F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A', B and B', or C and C', as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

   2. $\leq$ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

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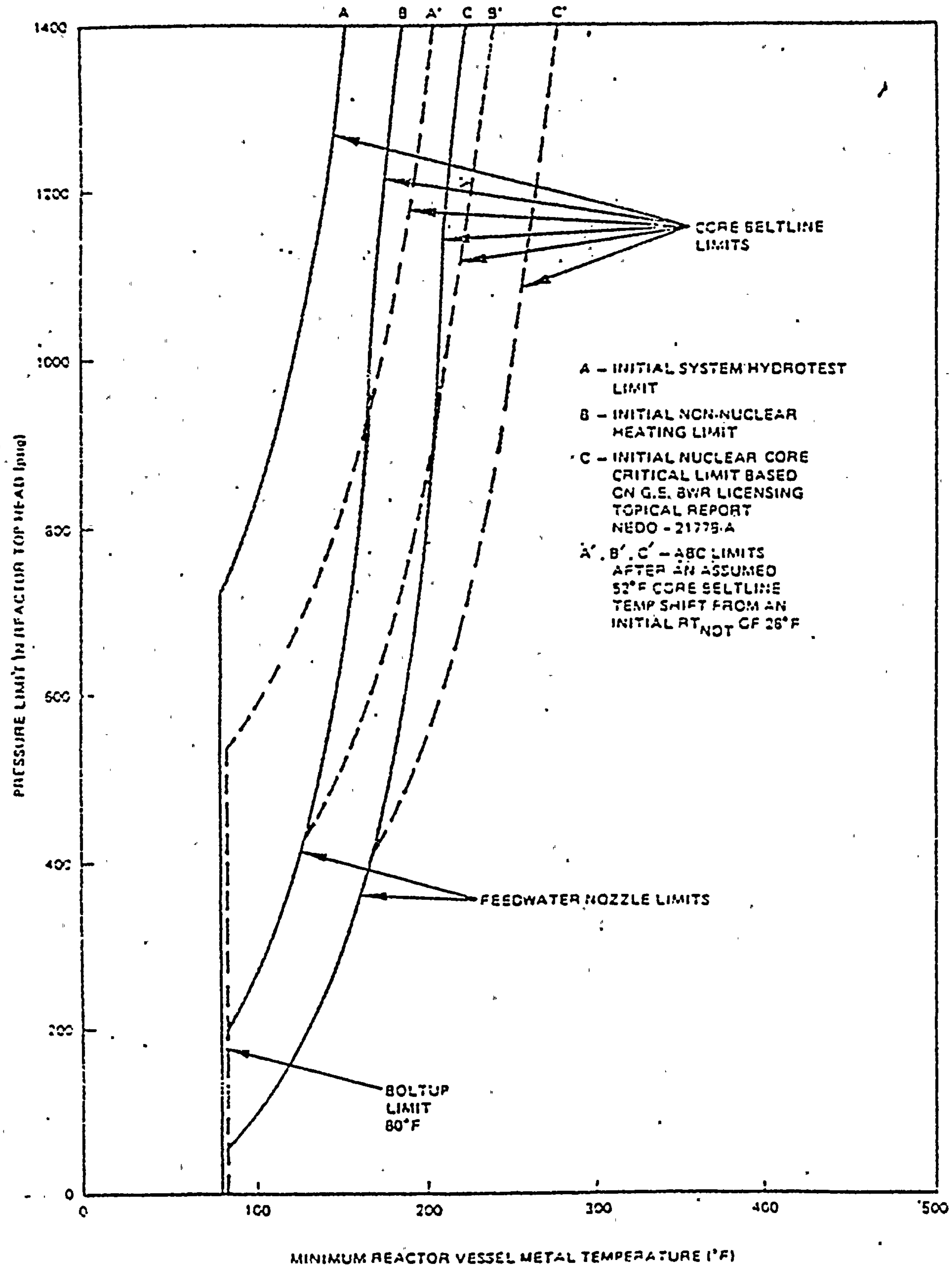

Figure 3.4.6.1-1 WPPSS Hanford Unit 2 Minimum Temperature Required Vs. Reactor Pressure

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE

Capsule Number	Vessel Location Azimuth	Lead Factor	Withdrawal Time (EFPY)
1	300°	Due to symmetry, all capsules are expected to have the same lead factor.	8
2	120°		24
3	30°	LF = 1.2 at the ¼T LF = 0.86 at vessel ID	Standby

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

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LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than (~~1045~~) 1020 psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding (~~1045~~) 1020 psig, reduce the pressure to less than (~~1045~~) 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than (~~1045~~) 1020 psig at least once per 12 hours.

*Not applicable during anticipated transients.

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REACTOR COOLANT SYSTEM

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to {3} and less than or equal to {5} seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

   a) Restore the inoperable valve(s) to OPERABLE status, or

   b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between {3} and {5} seconds when tested pursuant to Specification 4.0.5.

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3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

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

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5.

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3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.**

b. With no RHR shutdown cooling mode loop in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system or alternative method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

**Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system or alternative method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

##The shutdown cooling mode loop may be removed from operation during hydrostatic testing.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. Seven OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

2. Seven OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*,# and 3*.

---

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~100~~ 128 psig.

#See Special Test Exception 3.10.6.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:

1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

c. For ECCS division 3, provided that ECCS divisions 1 and 2 and the RCIC system are OPERABLE:

1) With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

2) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:

1) With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2) With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

e. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:

1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ ~~100~~ 128 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ ~~100~~ 128 psig within the next 24 hours.

f. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the useage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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4.5.1 ECCS division 1, 2 and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI and HPCS systems:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifing that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

b. Verifing that, when tested pursuant to Specification 4.0.5, each:

1. LPCS pump develops a flow of at least ~~[illegible]~~ 6350 gpm ~~against a test line pressure~~ at a total head greater than or equal to (~~[illegible]~~ 290 psi.) ~~psig.~~

2. LPCI pump develops a flow of at least ~~[illegible]~~ 7450 gpm ~~against a test line pressure~~ at a total head greater than or equal to (~~[illegible]~~ 122 psi.) ~~psig.~~

3. HPCS pump develops a flow of at least ~~[illegible]~~ 6350 gpm ~~against a test line pressure~~ at a total head greater than or equal to (~~[illegible]~~ 427 psi.) ~~psig.~~

c. For the LPCS, LPCI and HPCS systems, at least once per 18 months performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

d. For the HPCS system, at least once per 18 months, verifying that the suction is (automatically) transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal and on a suppression pool high water level signal.

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

SURVEILLANCE REQUIREMENTS (Continued)

e. For the ADS by:

1. At least once per 24 hours, by verifying that the accumulator backup compressed gas system ~~pressure~~ capacity in each bottle is ≥(~~2200~~ 257) ~~psig~~. standard cubic feet.

2. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

3. At least once per 18 months:

a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig* and observing that either:

1) The control valve or bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow.

c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying an alarm setpoint of (~~1258~~ 140) psig on decreasing pressure.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

The A/E has specified a Matheson size 1A or equiv. bottle rated at 2490 psig, with a 257 scf capacity. The bottles have not yet been purchased for WNP-2 such that a specific pressure can be identified. The plant surveillance procedures will provide the required pressure based on the 257 scf design capacity.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

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EMERGENCY CORE COOLING SYSTEMS

~~3/4 5.2 ECCS - SHUTDOWN~~

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

1. From the suppression chamber, or

2. When the suppression pool level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water, equivalent to a level of ~~(_)~~ 7'7" in each tank or 13'3" in a single CST.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

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3.5.2 e.2

The potential exists for valving a CST out of service and the alarm setpoint on each tank is such that 135 K gallons is provided in both (½ each). The 13'3" limit will be administratively controlled to provide 135K in a single tank.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS divisions shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

4.5.2.2 The HPCS system shall be determine OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.e.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a.. In OPERATIONAL CONDITION 1, 2 and 3 with ~~a contained water volume of at least 127,000 ft³, equivalent to a level of 30'9 3/4"~~ an operating level of 30'11 3/4" and operating range of ±2 inches (466' 3/4" to 466'4 3/4" ele.) with a narrow range scale of ±25".

b. In OPERATIONAL CONDITION 4 and 5* with ~~a contained water volume of at least ( ) ft³, equivalent to a level of ( )~~, except that the suppression chamber level may be less than the limit or may be drained provided that:

an operating level of 30'11 3/4" and operating range of ±2" (466 2 3/4" ele. ± 2 inches)

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least 135,000 available gallons of water, equivalent to a level of ~~( )%, and~~ 7'7" with both tanks available or 13'3" with one tank in service.

4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded {or being flooded from the suppression pool}, the spent fuel pool gates are removed {when the cavity is flooded}, and the water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

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SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to ~~as applicable:~~ 30'9 3/4" at least once per 24 hours.

~~a. 30' 9 3/4") at least once per 24 hours.~~

~~b. ( )' at least once per (12) hours.~~

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 34.7 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

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CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal ~~to~~:

~~1.~~ L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 34.7 psig.

2. L_t, ( ) percent ~~by~~ weight of the containment air per 24 hours at a reduced ~~pressure~~ of P_t, ( ) psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests when pressurized to P_a, 34.7 psig.

c. *Less than or equal to 11.5 scf per hour for any one main steam line isolation valve when tested at ~~P_t~~, 25.0 psig. differential pressure

d. A combined leakage rate of less than or equal to 1 gpm times the total number of ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 Pa, 38.2 psig.

should be subscript

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, ~~or 0.75 L_t, as applicable~~, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line isolation valve, or

d. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate~~(s)~~ to less than or equal to 0.75 L_a ~~or 0.75 L_t, as applicable~~, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steamline isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests to less than or equal to 0.60 L_a, and

*Exemption to Appendix "J" of 10 CFR 50.

- Recommend P_t be deleted. Must establish a correlation between L_a and L_t which industry is finding difficult to do to NRC satisfaction.
- P_t @ statement 3.6.1.2.c may be confused with P_t in 3.6.1.2.a.2 (if 3.6.1.2.a.2 retained)

ACTION (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve, and

d. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which pentrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 34.7 psig, ~~or P_t, ( ) psig,~~ during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, ~~or 0.75 L_t, as applicable,~~ the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, ~~or 0.75 L_t, as applicable,~~ a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, ~~or 0.75 L_t, as applicable,~~ at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a. ~~or 0.25 L_t, as applicable.~~

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 34.7 psig. ~~or P_t, ( ) psig, as applicable.~~

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d. Type B and C tests shall be conducted with gas at P_a, 34.7 psig,* at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves,

~~3. Penetrations using continuous leakage monitoring systems,~~

3 ~~4~~. Valves pressurized with fluid from a seal system,

4 ~~5~~. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

5 ~~6~~. Purge supply and exhaust isolation valves with resilient seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

delete → g. ~~Type B periodic tests are not required for penetrations continuously monitored by the Containment Penetration Pressurization System, provided the system is OPERABLE per Specification 3.6.1.9.~~

delete → h. ~~Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 34.7 psig, at intervals no greater than once per 3 years.~~

g ~~i~~. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 38.2 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h ~~j~~. ECCS and RCIC containment isolation valves in hydrostatically tested lines which pentrate the primary containment shall be leak tested at least once per 18 months.

i ~~k~~. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.8.~~a~~ 2 and 4.6.1.8.~~b~~ 3

j ~~l~~. The provisions of Specification 4.0.2 are not applicable to 24 month or 40 ± 10 month surveillance intervals.

*Unless a hydrostatic test is required per Table 3.6.3-1.

Justification for deleting 4.6.1.2.g, h.

In a conversation with R. Bottimore of the NRC, the meanings of "Containment Penetration Pressurization System" and of "continuous Leakage monitoring system" were clarified.

Pressurization systems prevent leakage from the penetration.

Monitoring systems would only measure leakage but not prevent it.

As currently designed, WNP-2 has no such systems.

DRAFT

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 34.7 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

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

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

DRAFT

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 0.025 L_a when the gap between the door seals is pressurized to 10 psig or greater.

b. By conducting an overall air lock leakage test at P_a, 34.7 psig and by verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months#, and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance had been performed on the air lock that could affect the air lock sealing capability*.

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.**

#The provisions of Specification 4.0.2 are not applicable.

*Exception to Appendix J of 10 CFR 50.

**Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been de-inerted.

CONTAINMENT SYSTEMS

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting the blower(~~s~~) from the control room and operating the blower(~~s~~) for at least 15 minutes.
2. Energizing the heaters and verifying ~~(a temperature rise of greater than or equal to ( )°F within ( ) minutes)~~ {current of ( ) amperes ± {10}% per phase for each heater}. LATER

b. During each COLD SHUTDOWN {, if not performed within the previous 92 days, ~~by cycling each (bleeder) valve and (steam isolation) valve through at least one complete cycle of full travel)~~ {in accordance with Specification 4.0.5}.

c. At least once per 18 months by:

1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and the blower starts.
2. Verifying that the blower develops at least the below required vacuum at the rated capacity:
   a) Inboard valves, (~~50~~) 20" H_2O at (~~100~~) 50 scfm.
   b) Outboard valves, (~~50~~) 20" H_2O at (~~240~~) 50 scfm.

d. By verifying the {flow, pressure and temperature} ~~{operating}~~ instrumentation to be OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
2. CHANNEL CALIBRATION at least once per 18 months.

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1. ACTION: INDICATED HEATER CURRENT AS METHOD OF VERIFYING HEATER OPERABILITY. SHOW CURRENT VALUE AS "LATER"

JUSTIFICATION

HEATER CURRENT IS AN ADEQUATE METHOD OF VARIFYING HEATER OPERABILITY, THE ACTUAL CURRENT VALUE WILL BE DETERMINED DURING PREOPERATIONAL TESTING.

2. ACTION: 4.6.1.4.b. REFERENCE SPECIFICATION 4.0.5 FOR OPERABILITY TESTING. DURING COLD SHUTDOWN.

JUSTIFICATION: THE MAIN STEAM LEAKAGE CONTROL VALVES ARE INCLUDED IN THE SUPPLY SYSTEM PUMP AND VALVE PROGRAM.

3. ACTION: 4.6.1.4.C.2 REVISED VALUES FOR VACUUM AND FLOW RATE OF SYSTEMS.

JUSTIFICATION: VALUES ARE PLANT SPECIFIC AND REFLECT THE SYSTEM DESIGN.

DRAFT

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistant with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the primary steel containment, the inspection procedure and the corrective actions taken.

DRAFT

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between ~~-2.0~~ -1.0 and ~~+2.0~~ +1.69 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell and suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

DRAFT

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the arithmetical average of the temperatures at a minimum of three of the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	504'0"	72°
b.	504'0"	190°
c.	504'0"	275°
d.	560'0"	355°
e.	547'0"	222°

Rewrite per next page

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

~~3.6.1.8 The drywell and suppression chamber 2 inch purge supply and exhaust isolation valves shall be OPERABLE and:~~

~~a. Each 24 and 30 inch purge supply and exhaust isolation valve shall be sealed closed.~~

~~b. Each 2 inch purge valve may be open for purge system operation for inerting, deinerting and pressure control.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

~~a. With a 24 and/or 30 inch drywell and suppression chamber purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal the 24 and 30 inch valve(s) or otherwise isolate the penetration within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

a. ~~b.~~ With a ~~2 inch~~ drywell and suppression chamber purge supply and/or exhaust isolation valve~~(s)~~ inoperable or open ~~(for more than (90) hours per 365 days)~~ for other than inerting, deinerting or pressure control, close the ~~open 2 inch~~ valve[s] or otherwise isolate the penetration~~(s)~~ within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~c.~~ With a drywell and suppression chamber purge supply and/or exhaust isolation valve~~(s)~~ with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirements ~~4.6.1.8.2~~ 4.6.1.8.1 ~~and/or 4.6.1.8.4~~ and/or 4.6.1.8.2, restore the inoperable valve[s] to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.]

SURVEILLANCE REQUIREMENTS

~~4.6.1.8.1 Each 24 and 30 inch drywell and suppression chamber purge supply and exhaust isolation valve shall be verified to be sealed closed at least once per 31 days.~~

4.6.1.8.1 ~~4.6.1.8.2~~ At least once per 6 months on a STAGGERED TEST BASIS each ~~sealed closed~~ 24 and 30 inch drywell and suppression chamber purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to [0.05] L_a when pressurized to P_a.]

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3.6.1.8 The drywell and suppression chamber purge supply and exhaust isolation valves shall be OPERABLE, with their use limited to inerting, de-inerting, and containment pressure control.

Justification for Change:

Since the bases for this item has changed, the item must change to accomodate the new bases. That is; the subject valves have all been demonstrated capable of closing during a LOCA or steam line break accident and therefore do not need to be "sealed" closed. (See Supply System Letter G02-83-170 dated Feb. 24, 1983 to Mr. A. Schwencer, Chief; Licensing Branch No. 2, Wash. D.C.) The new version of 3.6.1.8 above is in direct compliance with the WNP-2 SER paragraph 6.2.4.3.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.8.2

~~4.6.1.8.2~~ At least once per 92 days each 2 inch drywell and suppression chamber purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to ≶0.01≷ L_a when pressurized to P_a.≷

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DELETE

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT PENETRATION PRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

~~3.6.1.9 The primary containment penetration pressurization system shall be OPERABLE.~~

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

~~ACTION:~~

~~With the primary containment penetration pressurization system inoperable, restore the system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.6.1.9 The primary containment penetration pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a, 38.2 psig, and has adequate capacity to maintain system pressure for at least 30 days.~~

This LCO is to be deleted. WNP-2 has elected to delete the system and will perform 10 CFR 50 Appendix J type B testing as required in LCO 3.6.1.2.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. ~~Volume between 127,800 ft³ and 128,300 ft³, equivalent to a~~ Level above ~~between 30' 9 3/4" and 31' 1 3/4", and a~~ 466' 0 3/4" ele, and a
2. Maximum average temperature of 90°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:
   a) 105°F during testing which adds heat to the suppression chamber.
   b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.
   c) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{K}$ design value of 0.05 ft².

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than 90°F, restore the average temperature to less than or equal to 90°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 90°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With the suppression chamber average water temperature greater than:
   a) 90°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
   b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.
3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

3.6.2.1.a.1

The volume information is provided in the bases and unnecessarily confusing if added here.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water temperature instrumentation channel in any sector inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression chamber water temperature to be within the limits at least once per 12 hours.

d. With more than one suppression pool water temperature instrumentation channel in the same sector inoperable, restore at least one inoperable water temperature instrumentation channel in each sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 90°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to 90°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER after suppression chamber average water temperature has exceeded 90°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 90°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least ~~eight~~ suppression pool water temperature instrumentation channels, at least one in each suppression pool sector, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the water high temperature alarm setpoints for 90°F & 105°F ~~≤ ( ) °F~~.

* d. At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 1.5 ~~(5)~~ psi and verifying that the A/√K calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

* The Supply System is to perform tests at the following frequency:

a. Preoperational tests at 1.5 and 5.0 psid,
b. At the first refueling, a test at 1.5 psid, and
c. At the first (40 month) ILRT interval, test at 1.5 and 5.0 psid.

A special report will then be submitted to the NRC-CSB, based on the information obtained in a. through c. above, to request discontinuation of further testing at 5.0 psid. Upon approval, all subsequent tests will be conducted at 1.5 psid on an 18-month frequency.

Justification:

d) The format was retained, changing only the 5.0 to 1.5 psi, which is the target testing pressure and format . Upon NRC approval of the special report findings, the paragraph will remain unchanged and require, but not necessitate, removal of the * note.

SUPPRESSION POOL AND DRYWELL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool and drywell spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHRSW heat exchangers and the suppression pool and dryw~~a~~ell spray spargers.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool and/or drywell spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool and/or drywell spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool and drywell spray mode of the RHR system shall be demonstrated OPERABLE:

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

b. By verifying that each of the required RHR pumps develops a flow of at least ~~(300)~~ 450 gpm on recirculation flow through the RHR heat exchanger and suppression pool spray sparger when tested pursuant to Specification 4.0.5.

c. By performance of an air or smoke flow test of the drywell spray nozzles at least once per 5 years and verifying that each spray nozzle is unobstructed.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump; and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHRSW heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

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

b. By verifying that each of the required RHR pumps develops a flow of at least 7450 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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CONTAINMENT SYSTEMS

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: ~~As shown in Table 3.6.3-1.~~ OPERATIONAL CONDITION 1, 2 and

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

1. Restore the inoperable valve(s) to OPERABLE status, or
2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or
3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

Otherwise, ~~in OPERATIONAL CONDITION 1, 2 or 3,~~ be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

delete ~~Otherwise, in Operational Condition,** suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.~~

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either;

1. The inoperable valve is returned to OPERABLE status, or
2. The instrument line is isolated and the associated instrument is declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

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

delete

1. Primary Containment Integrity is neither necessary nor sufficient when handling fuel in Secondary Containment. Other Technical Specifications require that Secondary Containment Integrity be maintained and that Standby Gas Treatment be operable during such operations. This is sufficient to prevent off site releases. (These requirements are in addition to minimum fuel pool water level specifications.)

2. In modes 4 and 5, redundant ECCS systems or subsystems are required unless the cavity is flooded, open to the fuel pool and has proper level. Additionally, during CORE ALTERATIONS, Secondary Containment Integrity and SGT operability is required. The requirements for limiting loss of vessel inventory and off site releases during OPERATIONAL CONDITION ** are adequately addressed elsewhere in Technical Specifications (ie TECH SPECS 3/4.5.2 and 3/4.6.5)

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CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow at greater than a 10 psid differential pressure in hydraulic service and 15 psid differential pressure in pneumatic service.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

{b. At least once per 18 months by removing ~~(at least one)~~ {the} explosive squib~~(s)~~ from {at least one} ~~(the)~~ explosive valve~~(s)~~ ~~(,~~ such that the ~~each~~ explosive squib in each explosive valve will be tested at least once per 90 ~~36~~ months~~,)~~ and initiating the explosive squib~~(s)~~. The replacement charge for the exploded squib~~(s)~~ shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable}.

The 90 month frequency allows for testing 5 squibbs on an 18 month interval. The commitment is consistent with communications with the containment systems branch and is consistent w/ LaSalle.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a) ~~(APPLICABLE OPERATIONAL CONDITIONS)~~	MAXIMUM ISOLATION TIME (Seconds)
a. Automatic Isolation Valves		
MAIN STEAM Iso. Valves	1	5*
MS-V-22A,B,C,D (b)		
MS-V-28A,B,C,D (b)		
Main Steam Line Drains	1	
MS-V-16		15
MS-V-19		15
MS-V-67A,B,C,D (b)		5
Reactor Recirc. Cooling Sample Values	2	5
RRC-V-19		
RRC-V-20		
Containment Purge Exhaust and Supply #	3	
CEP-V-1A,2A,3A,4A		4
CEP-V-1B,2B,3B,4B		1
CSP-V-1		4
CSP-V-2		4
CSP-V-3		4
CSP-V-4		4

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP	~~(APPLICABLE OPERATIONAL CONDITIONS)~~	MAXIMUM ISOLATION TIME (Seconds)
a. **Automatic Isolation Valves**			
Equipment Drain (Radioactive)		4	15
EDR-V-19			
EDR-V-20			
Floor Drain (Radioactive)		4	15
FDR-V-3			
FDR-V-4			
Fuel Pool Cooling/Suppression Pool Cleanup		4	30
FPC-V-153[f]			
FPC-V-154[f]			
FPC-V-156			
Reactor Recirculation Hydraulic Control[(e)]		4	5
HY-V-17A,B			
HY-V-18A,B			
HY-V-19A,B			
HY-V-20A,B			

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TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP	~~(APPLICABLE OPERATIONAL CONDITIONS)~~	MAXIMUM ISOLATION TIME (Seconds)
a. **Automatic Isolation Valves**			
Reactor Recirculation Hydraulic Control (cont'd)[e]		4	5
HY-V-33A,B			
HY-V-34A,B			
HY-V-35A,B			
HY-V-36A,B			
Traversing Incore Probe Valve		4	NA
TIP-V-1,2,3,4,5			
Reactor Closed Cooling		4	50
RCC-V-5			
RCC-V-21			
RCC-V-40			
RCC-V-104			
Radiation Monitoring Supply and Return		4	5
PI-VX-250			
PI-VX-251			
PI-VX-253			
PI-VX-256			
PI-VX-257			
PI-VX-259			

VERIFY still w/ PASS

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	~~VALVE GROUP~~ (APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)
a. **Automatic Isolation Valves**		
Residual Heat Removal		
RHR-V-123 A,B(g)(b)	5	5
RHR-V-8(g)(b)	6	40
RHR-V-9(g)(b)	6	40
RHR-V-23(g)(b)	6	90
RHR-V-53A,B(g)(b)	6	40
RHR-V-11 A,B	10	20
RHR-V-24A,B(c)	10	270
RHR-V-21	10	270
RHR-V-27A,B(c)	10	30
Reactor Water Cleanup System	7	30
RWCU-V-1(d)		
RWCU-V-4		

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP	~~(APPLICABLE OPERATIONAL CONDITIONS)~~	MAXIMUM ISOLATION TIME (Seconds)
a. **Automatic Isolation Valves**			
Reactor Core Isolation Cooling			
RCIC-V-8		8	20
RCIC-V-63		8	16
RCIC-V-64		12	16
RCIC-V-76		8	5
Low Pressure Core Spray			
LPCS-V-12		10	180
High Pressure Core Spray			
HPCS-V-23		11	180

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3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division 1, consisting of:

1. 125 volt battery B1-1.
2. 250 volt battery B2-1.
3. ±24 volt batteries B0-1A and B0-1B.
4. 125 volt full capacity charger C1-1.
5. 250 volt full capacity charger C2-1.
6. ±24 volt full capacity chargers C0-1A and C0-1B.

b. Division 2, consisting of:

1. 125 volt battery B1-2.
2. ±24 volt batteries B0-2A and B0-2B.
3. 125 volt full capacity charger C1-2.
4. ±24 volt full capacity chargers C0-2A and C0-2B.

c. Division 3, consisting of:

1. 125 volt battery B1-HPCS.
2. 125 volt full capacity charger C1-HPCS.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division 1 or Division 2 battery ~~and~~ or charger of the above required D.C. electrical power sources inoperable, restore the inoperable division battery to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division 3 battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

"and/or" is grammatically incorrect since if "or" is applied, the "and" would never be applied unless their was a double failure of a battery bank and a charger. The or would apply for the "and" case as well.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 250-volt 125-volt ±24-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

2. Total battery terminal voltage on float charge is greater than or equal to 25.8-volts, 129-volts and 258-volts for the ±24-volt, 125-volt and 250-volt batteries, respectively.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below [22]-volts, [110]-volts and [220]-volts for the ±24-volt, 125-volt and 250-volt batteries, respectively, or battery overcharge with battery terminal voltage above [31]-volts, [150]-volts and [300]-volts for the ±24-volt, 125-volt and 250-volt batteries, respectively, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 250×10^{-6} ohms, and

3. The average electrolyte temperature of [a representative number] of connected cells is above (60°F).

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The resistance of each cell[-to-cell] and terminal connection is less than or equal to 250×10^{-6} ohms, and

4. The battery charger will supply:

    1. For ±24-volt batteries at least 25 amperes at a minimum of 25.8 volts for at least [4] hours.

    2. For the 125-volt batteries, at least 200 amperes at a minimum of 129 volts for at least [4] hours.

    3. For the 250-volt battery, at least 400 amperes at a minimum of 258 volts for at least [4] hours.

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48.2.1.6. LOW BATTERY TERMINAL VOLTAGE BASED ON 1.7 Volts/CELL.
CUTOFF VOLTAGE BASED ON 2.57 VOLTS/CELL.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for {8} hours for Divisions 1 and 2 and {8} hours for Division 3 when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to ~~( ) volts~~ 21 VOLTS, 10C VOLTS, 203 VOLTS, FOR THE ±24 VOLT, 125 VOLT, & 250 VOLT BATTERIES, RESPECTIVELY.

   a) ±24 VOLT BATTERY ~~Battery (1A)~~, greater than or equal to (60) amperes; 125 VOLT battery ~~{1B}~~, greater than or equal to {222} amperes; and 250 battery ~~{1C}~~, greater than or equal to {612} amperes during the initial 60 seconds of the test.

   b) ±24 VOLT Battery ~~{1A}~~, greater than or equal to (60) amperes; 125 VOLT battery ~~{1B}~~, greater than or equal to {101} amperes; and 250 VOLT battery ~~{1C}~~, greater than or equal to {432} amperes during the remainder of the first hour of the test.

   c) ±24 VOLT Battery ~~{1A}~~, greater than or equal to (60) amperes; 125 VOLT battery ~~{1B}~~, greater than or equal to {101} amperes; and 250 VOLT battery ~~{1C}~~, greater than or equal to {229} amperes during the remainder of the {8} hour test.

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. At this once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months during shutdown performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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4.8.2.1.d.2 LOWER VOLTAGE LIMIT BASED ON IEEE-450-1975 RECOMMENDATION

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195 Average of all connected cells > 1.205	Not more than .020 below the average of all connected cells Average of all connected cells ≥ 1.195[b]

[a] Corrected for electrolyte temperature and level.

[b] Or battery charging current is less than (2) amperes when on float charge.

[c] May be corrected for average electrolyte temperature.

[1] For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

[2] For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

[3] Any Category B parameter not within its allowable value indicates an inoperable battery.

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ELECTRICAL POWER SYSTEMS

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division 1 or Division 2, and, when the HPCS system is required to be OPERABLE, Division 3, of the D.C. electrical power sources shall be OPERABLE with:

a. Division 1, consisting of:
   1. 125 volt battery B1-1.
   2. 250 volt battery B2-1.
   3. ±24 volt batteries B0-1A and B0-1B.
   4. 125 volt full capacity charger C1-1.
   5. 250 volt full capacity charger C2-1.
   6. ±24 volt full capacity chargers C0-1A and C0-1B.

b. Division 2, consisting of:
   1. 125 volt battery B1-2.
   2. ±24 volt batteries B0-2A and B0-2B.
   3. 125 volt full capacity charger C1-2.
   4. ±24 volt full capacity chargers C0-2A and C0-2B.

c. Division 3, consisting of:
   1. 125 volt battery B1-HPCS.
   2. 125 volt full capacity charger C1-HPCS.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the Division 1 ~~and~~/or Division 2 battery ~~and~~/or charger of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With Division 3 battery ~~and~~/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

*When handling irradiated fuel in the secondary containment.

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ELECTRICAL POWER SYSTEMS

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~~3/4.8.3~~ ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized with tie breakers open ~~(both)~~ between redundant buses within the unit ~~(and between units at the same station)~~:

N/A for WNP-2

a. A.C. Power Distribution

1. Division 1, consisting of:
   a) 4160 volt bus SM-7.
   b) 480 volt bus SL-71 and SL-73.
   c) 480 volt MCC's 7A, 7A-A, 7B, 7B-A, 7B-B, 7F.
   d) 480 volt Power Panel PP-7A-B.
   e) 120/208 volt 3Ø Power Panels PP-7A-G, PP-7A-A-A.
   f) 120/240 volt 1Ø Power Panels PP-7A-A, PP-7A-F, PP-7A-E, and PP-7A.

2. Division 2, consisting of:
   a) 4160 volt bus SM-8.
   b) 480 volt bus SL-81 and SL-~~8X~~ 83.
   c) 480 volt MCC's 8A, 8A-A, 8B, 8B-A, 8B-B, 8F.
   d) 480 volt Power Panel PP-8A-B.
   e) 120/208 volt 3Ø Power Panels PP-8A-G, PP-8A-A-A.
   f) 120/240 volt 1Ø Power Panels PP-8A-A, PP-8A-F, PP-8A-E, and PP-8A.

3. Division 3, consisting of:
   a) 4160 volt bus SM-4.
   b) 480 volt 3Ø Engine & Gen. Aux. loads Power Panel.
   c) 120/240 volt 1Ø Power Panel PP-4A.
   d) 480 volt 3Ø MCC 4A

b. D.C. Power Distribution

1. Division 1, consisting of:
   a) 125 volt D.C. Main Distribution Panel S1-1.
   b) 125 volt VDC Motor Control Center MC-S1-1D.
   c) 125 VDC Instr. and Control NSSS Bd. Distr. Panel DP-S1-1A.
   d) 125 VDC Critical Swgs. & Remote Shutdn. Distr. Pnl. DP-S1-1D.
   e) 125 VDC Diesel Gen. 1 Dist. Pnl. DP-S1-1E.
   f) 250 VDC Main Distribution Panel S2-1.
   g) 250 VDC Motor Control Center MC-S2-1A, Part A and Part B.
   h) ±24 VDC Power Panel DP-S0-A.

2. Division 2, consisting of:
   a) 125 volt D.C. Main Distribution Panel S1-2.
   b) 125 volt VDC Motor Control Center MC-S1-2D.
   c) 125 VDC Instr. and Control NSSS Distr. Panel DP-S1-2A.
   d) 125 VDC Critical Swgs. & Remote Shutdn. Distr. Pnl. DP-S1-2D.
   e) 125 VDC Diesel Gen. 2 Dist. Pnl. DP-S1-2E.
   f) ±24 VDC Power Panel DP-S0-B.

3. Division 3, consisting of 125 volt D.C. HPCS distribution panel.

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LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. For A.C. power distribution:

1. With either Division 1 or Division 2 of the above required A.C. distribution system not energized, re-energize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division 3 of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

b. For D.C. power distribution:

1. With either Division 1 or Division 2 of the above required D.C. distribution system not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division 3 of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the ~~buses/MCCs/panels.~~ medium voltage buses.

Justification (to include change in 4.8.3.2):

Lower voltage mcc's and power panels do not have voltage indication.

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LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following power distribution system divisions shall be energized:

a. For A.C. power distribution, Division 1 or Division 2, and when the HPCS system is required to be OPERABLE, Division 3, with:

1. Division 1, consisting of:
   a) 4160 volt bus SM-7.
   b) 480 volt bus SL-71 and SL-73.
   c) 480 volt MCC's 7A, 7A-A, 7B, 7B-A, 7B-B, 7F.
   d) 480 volt Power Panel PP-7A-B.
   e) 120/208 volt 3Ø Power Panels PP-7A-G, PP-7A-A-A.
   f) 120/240 volt 1Ø Power Panels PP-7A-A, PP-7A-F, PP-7A-E, and PP-7A.
2. Division 2, consisting of:
   a) 4160 volt bus SM-8.
   b) 480 volt bus SL-81 and SL-~~93~~ 83
   c) 480 volt MCC's 8A, 8A-A, 8B, 8B-A, 8B-B, 8F.
   d) 480 volt Power Panel PP-8A-B.
   e) 120/208 volt 3Ø Power Panels PP-8A-G, PP-8A-A-A.
   f) 120/240 volt 1Ø Power Panels PP-8A-A, PP-8A-F, PP-8A-E, and PP-8A.
3. Division 3, consisting of:
   a) 4160 volt bus SM-4.
   b) 480 volt 3Ø Engine & Gen. Aux. loads Power Panel.
   c) 120/240 volt 1Ø Power Panel PP-4A.
   d) 480 volt 3Ø MCC 4A

b. For D.C. power distribution, Division 1 or Division 2, and when the HPCS system is required to be OPERABLE, Division 3, with:

1. Division 1, consisting of:
   a) 125 volt D.C. Main Distribution Panel S1-1.
   b) 125 volt VDC Motor Control Center MC-S1-1D.
   c) 125 VDC Instr. and Control NSSS Bd. Distr. Panel DP-S1-1A.
   d) 125 VDC Critical Swgs. & Remote Shutdn. Distr. Pnl. DP-S1-1D.
   e) 125 VDC Diesel Gen. 1 Dist. Pnl. DP-S1-1E.
   f) 250 VDC Main Distribution Panel S2-1.
   g) 250 VDC Motor Control Center MC-S2-1A, Part A and Part B.
   h) ±24 VDC Power Panel DP-S0-A.
2. Division 2, consisting of:
   a) 125 volt D.C. Main Distribution Panel S1-2.
   b) 125 volt VDC Motor Control Center MC-S1-2D.
   c) 125 VDC Instr. and Control NSSS Distr. Panel DP-S1-2A.
   d) 125 VDC Critical Swgs. & Remote Shutdn. Distr. Pnl. DP-S1-2D.
   e) 125 VDC Diesel Gen. 2 Dist. Pnl. DP-S1-2E.
   f) ±24 VDC Power Panel DP-S0-B.
3. Division 3, consisting of 125 volt D.C. HPCS distribution panel.

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Excess Flow Check Valves (e)

Containment Atmosphere

PI-EFC-X29 b/d	PI-EFC-X61c
" " -X29 e/f	" " -X62b
" " -X30 a/c.	" " -X69 d/f
" " -X30 d/f	" " -X72b
" " -X42 e/f	" " -X73C

Reactor Pressure Vessel

PI-EFC-X2

-X12A,B,C

-X18 A,B,C,D

-X37e,f

-X38 a,b,c,d,e,f

-X39 a,b,d,e.

-X40 c,d

-X41 c,d

-X42 a,b

-X44A a,b,c,d,e,f,g,h,j,k,l,m

-X44B a,b,c,d,e,f,g,h,j,k,l,m

-X61 a,b

-X62c,d

-X66

-X67

-X69 a,b,e

Reactor Pressure Vessel (cont'd)

PI-EFC-X70a, b, c, d, e, f

-X71 a, b, c, d, e, f

-X72a

-X73a

-X74 a, b, e, f

-X75 a, b, c, d, e, f

-X78 b, c, f

-X79 a, b

-X82 b

-X84a

-X106

-X107

-X108

-X109

-X110

-X111

-X112

-X113

-X114

-X115

-X119.

Other

PI-EFC-X40e, f

-X41e, f

-X86A, B

-X87A, B

Manual Containment Isolation Valves

Demineralized Water

DW -V-156
DW -V-157

Containment Air System

CAS-VX-82E

Service Air

SA-V-109

Fuel Pool / Suppression Pool Cooling

FPC-V-149

Residual Heat Removal
RHR-V-121

Reactor Core Isolation Cooling
RCIC-V-742

Air Supply to Testable Check Valves

Air Supply	Check Vv
PI-VX-42d PI-VX-216	RHR-V-50A
PI-VX-69C PI-VX-221	RHR-V-50B
PI-VX-61f PI-VX-219	RHR-V-41A
PI-VX-54Bf PI-VX-218	RHR-V-41B
PI-VX-62f PI-VX-220	RHR-V-41C
LPCS-V-66 LPCS-V-67	LPCS-V-6
HPCS-V-65 HPCS-V-68	HPCS-V-5
RCIC-V-184 RCIC-V-740	RCIC-V-66

Other Containment Isolation Valves.

	Maximum Isolation time (seconds)
Main Steam Leakage Control[b]	
MSLC-V-3A,B,C,D	≤ 8
Reactor Feedwater/ RWCU Return	
RFW-V-65 A,B	≤ 120
RFW-V-10 A,B	N/A
RFW-V-32 A,B	N/A
RWCU-V-40	≤ 30
High Pressure Core Spray	
HPCS-V-5[(g)(b)]	N/A
HPCS-V-4[(g)(b)]	≤ 7
HPCS-V-15[(...)]	≤ 8
HPCS-V-12	≤ 4
HPCS-RV-14[(e)(h)]	N/A
HPCS-RV-35[(e)(h)]	N/A
Low Pressure Core Spray	
Ispace LPCS-V-1[(f)(h)]	≤ 120
LPCS-V-5[(g)(h)]	≤ 27
LPCS-V-6[(g)(h)]	N/A
LPCS-FCV-11	≤ 45
LPCS-RV-31[(e)(h)]	N/A
LPCS-RV-15[(e)(h)]	N/A
Standby Liquid Control	
SLC-V-7	N/A
SLC-V-4A,B	N/A

Reactor Core Isolation Cooling	Maximum Isolation Time (Seconds)
RCIC-V-19	≤ 5
RCIC-V-68	≤ 50
RCIC-V-40	N/A
RCIC-V-69	≤ 8
RCIC-V-28	N/A
RCIC-V-31 (f)(h)	≤ 40
RCIC-V-66 (g)(h)	N/A
RCIC-V-13 (g)(h)	≤ 15

Residual Heat Removal / Low Pressure Injection	
RHR-V-16A, B	≤ 10
RHR-V-17A, B	≤ 10
RHR-V-41A, B, C (g)(h)	N/A
RHR-V-42A, B, C (g)(h)	≤ 27
RHR-V-50A, B (g)(h)	N/A
RHR-V-209 (g)(h)	N/A
RHR-RV-25A, B, C (e)(h)	N/A
RHR-RV-55A, B (e)(h)	N/A
RHR-RV-36 (e)(h)	N/A
RHR-FCV-64A, B, C (e)(h)	≤ 15
RHR-RV-1A, B	N/A
RHR-V-73A, B	≤ 20
RHR-V-120	N/A
RHR-V-134A, B	≤ 10
RHR-RV-88A, B, C (e)(h)	N/A
RHR-RV-5 [illegible]	N/A
RHR-RV-30 (e)(h)	N/A
RHR-V-4A, B, C (f)(h)	≤ 120
RHR-RV-95A, B (e)(h)	N/A
RHR-V-124A, B	≤ 8
RHR-V-125A, B	≤ 8
RHR-V-101A, B	
RHR-V-102A, B	

Containment Atmosphere Control [H$_2$ Recombiner] (1)

	Maximum Isolation Time (Seconds)
CAC-V-2	≤ 20
CAC-FCV-2A, B,	≤ 38
CAC-V-15	≤ 20
CAC-FCV-1A, B,	≤ 38
CAC-V-11.	≤ 20
CAC-V-6	≤ 20
CAC-V-4	≤ 20
CAC-FCV-4A, B	≤ 38
CAC-V-13	≤ 20
CAC-V-17	≤ 20
CAC-FCV-3A, B	≤ 38
CAC-V-8	≤ 20

Containment Purge System

CSP-V-5	≤ 4
CSP-V-6	≤ 4
CSP-V-7	N/A
CSP-V-8	N/A
CSP-V-9	≤ 4
CSP-V-10	N/A

Reactor Recirculation (Seal Injection)

RRC-V-13A, B	N/A
RRC-V-16A, B	≤ 4

Containment Instrument Air

CIA-V-20	≤ 12
CIA-V-21	N/A
CIA-V-30A, B	≤ 8
CIA-V-31A, B	N/A

Containment Air Supply

CAS-V-453

Post Accident Sampling System (c)

PSR-V-X73-1
PSR-V-X73-2
PSR-V-X77A1
PSR-V-X77A2
PSR-V-X77A3
PSR-V-X77A4
PSR-V-X80-1
PSR-V-X80-2
PSR-V-X82-1
PSR-V-X82-2

PSR-V-X82-7
PSR-V-X82-8
PSR-V-X83-1
PSR-V-X83-2
PSR-V-X84-1
PSR-V-X84-2
PSR-V-X88-1
PSR-V-X88-2

Footnotes

* But greater than 3-seconds.

\# Provisions of Technical Specification 3.0.4 are not applicable.

(a) See Technical Specification 3.3.2 for the isolation signal(s) which operate each group.

(b) Valve leakage not included in sum of Type B and C tests

(c) May be opened on an intermittent basis under administrative control.

(d) Not closed by SLC actuation signal

(e) Not subject to Type C Leak Rate Test

(f) Hydraulic leak test at 35.2 psig

(g) Not subject to Type C test. Test per Technical Specification 4.4.3.2.d.

(h) Tested as part of Type A test.

(i) May be tested as part of Type A test. If so tested, Type C test results may be excluded from sum of other Type B and C tests.

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3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Each pair of suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

replace w/ attached

a. With one or more vacuum breakers in one pair of suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pair of vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d ~~c~~. With {one} ~~(the)~~ position indicator of any suppression chamber - drywell vacuum breaker inoperable:

1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, ~~(and)~~ {or}

{2. Verify the vacuum breaker{s} with the inoperable position indicator to be closed by {conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter{.}

3. {Otherwise,} be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

a. With one suppression chamber-drywell vacuum breaker inoperable for opening but known to be closed, operations may continue provided Surveillance Requirement 4.6.4.1.b.1 is performed on the OPERABLE vacuum breakers within 2 hours and at least once per 15 days thereafter. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With more than one vacuum breaker inoperable for opening but known to be closed, restore the vacuum breakers such that at least eight pairs of vacuum breakers are OPERABLE within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one suppression chamber-drywell vacuum breaker open, operation may continue provided that the other vacuum breaker in the pair is verified closed:

1) By position indication within 2 hours and at least once per 15 days thereafter; or

2) By conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for one hour ~~without makeup~~ within 24 hours and at least once per 15 days thereafter.

3) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ARGUMENTS

a and b Page 5 of the technical report titled "Parametric Studies of Containment Negative Pressure Transients" supports operation with one vacuum breaker inoperable.

c Since we have a pair of vacuum breakers in series, if one of a pair is open, we should be able to use either of two methods to show the other valve closed; i.e., by position indication or by leak rate test across the pair of valves similar to the case in "d" for loss of indication of a closed valve.

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying {both} (~~the~~) position indicator{s} OPERABLE by observing expected valve movement during the cycling test.

3. At least once per 18 months by;

a) Verifying the force required to open each vacuum breaker does ~~opening setpoint, from the closed position, to be less than or equal to~~ not exceed the equivalent of {0.5} psid, and

b) Verifying {both} (~~the~~) position indicator{s} OPERABLE by performance of a CHANNEL CALIBRATION.

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REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 All Reactor Building - suppression chamber vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one Reactor Building - suppression chamber vacuum breaker inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one Reactor Building - suppression chamber vacuum breaker open, verify the other vacuum breaker in the line to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With {one} ~~(the)~~ position indicator of any Reactor Building - suppression chamber vacuum breaker inoperable, {restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker to be closed at least once per 24 hours by visual inspection. Otherwise,} {declare the vacuum breaker inoperable or} be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Reactor Building - suppression chamber vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days by:

   a) Cycling each vacuum breaker through at least one ~~([illegible])~~ {test} cycle. ~~(of full travel)~~.

   b) Verifying {both} ~~(the)~~ position indicator{s} OPERABLE by observing expected valve movement during the cycling test.

2. At least once per 18 months by:

   a) Demonstrating that the force required to open each vacuum breaker {CSP-V-7, CSP-V-8, + CSP-V-10} does not exceed the equivalent of 0.5 psid.

   b) Visual inspection.

   c) Verifying {both} ~~(the)~~ position indicator{s} OPERABLE by performance of a CHANNEL CALIBRATION.

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3. By demonstrating the vacuum breaker actuation instrumentation OPERABLE by performance of a:

   a) CHANNEL CHECK at least once per 24 hours.

   b) CHANNEL FUNCTIONAL TEST at least once per 31 days.

   c) CHANNEL CALIBRATION at least once per 18 months.

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3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inches of vacuum water gauge.

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatches and blowout panels are closed and sealed.

2. At least one ~~The~~ door in each access to the secondary containment is closed ~~, except for routine entry and exit.~~

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 120 seconds, and

2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate not exceeding 2240 CFM.

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

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CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT AUTOMATIC ISOLATION ~~(DAMPERS)~~ {VALVES}

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment ventilation system automatic isolation ~~(dampers)~~ {valves} shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation ~~(dampers)~~ {valves} shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation ~~(damper)~~ {valve} OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable ~~(damper)~~ {valve}(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated ~~(damper)~~ {valve} secured in the isolation position, or

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

Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ventilation system automatic isolation ~~(damper)~~ {valve} shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the ~~(damper)~~ {valve} to service after maintenance, repair or replacement work is performed on the ~~(damper)~~ {valve} or its associated actuator, control or power circuit by cycling the ~~(damper)~~ {valve} through at least one complete cycle of full travel and verifying the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation ~~(damper)~~ {valve} actuates to its isolation position.

c. AT LEAST ONCE PER 92 DAYS By verifying the isolation time to be within its limit ~~when tested pursuant to Specification 4.6.5.~~

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

CHANGE: ADDED THE FREQUENCY OF VALVE ISOLATION TIME TEST AND DELETED REFERENCE TO SPECIFICATION 4.0.5.

JUSTIFICATION: THE SUBJECT VALVES WERE DESIGNED AND BUILT TO ASME SECTION III REQUIREMENTS BUT THE ASME SECTION XI (4.0.5) PROGRAM IS APPLICABLE TO STEAM, LIQUID, AND RADIOACTIVE WASTE SYSTEMS AND THIS IS A VENTILATION SYSTEM. WE ARE VERIFYING THE ISOLATION TIMES ON A QUARTERLY BASIS WHICH WE FEEL IS ADEQUATE.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM-AUTOMATIC ISOLATION (DAMPERS)(VALVES)

DAMPER (VALVE) FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Supply ~~(Damper)~~ {Valve} ROA-V-1	4
2. Reactor Building Ventilation Supply ~~(Damper)~~ {Valve} ROA-V-2	4
3. Reactor Building Ventilation Exhaust ~~(Damper)~~ {Valve} REA-V-1	4
4. Reactor Building Ventilation Exhaust ~~(Damper)~~ {Valve} REA-V-2	4

(*The provisions of Specification 3.0.4 are not applicable.)

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LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

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

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than (~~*~~)% .05 and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ~~4400~~ 4457 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than (~~*~~ .05)%; and

3. Verifying a subsystem flow rate of ~~4400~~ 4457 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than (~~*~~)%. .05

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than (8) inches Water Gauge while operating the filter train at a flow rate of 4400 cfm ± 10%.

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Manual initiation from the control room, and

   b. Simulated automatic initiation signal.

3. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

4. Verifying that the heaters dissipate 20.7 ± 2.1 kw when tested in accordance with ANSI N510-1975.

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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 bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than ~~(*)~~ .05% in accordance with ANSI N510-1975 while operating the system at a flow rate of ~~4400~~ 4457 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than ~~(*)~~ .05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of ~~4400~~ 4457 cfm ± 10%.

~~(*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 99% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).)~~

~~(**0.175% value applicable when a charcoal adsorber efficiency of 99% is assumed, or 1% value applicable when a charcoal adsorber efficiency of 95% is assumed, or 10% value applicable when a charcoal adsorber efficiency of 90% is assumed in the NRC staff's safety evaluation.)~~

HEPA filter & charcoal adsorber removal capabilities consistent w/ FSAR analysis, chapter 6.5.

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3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system warmup test that the minimum recombiner outlet temperature increases to greater than or equal to ~~400°F~~ 500°F within 90 minutes.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner {operating} instrumentation and control circuits.

2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

c. At least once per 3 years by

~~3.~~ Verifying during a recombiner system functional test that, upon introduction of 1% by volume hydrogen in a 140-180 scfm stream containing at least 1% by volume oxygen, that the catalyst bed temperature rises in excess of 120°F within 20 minutes.

~~4.~~ 3. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e, loose wiring or structural connections, deposits of foreign materials. ~~etc.~~

d. By measuring the system leakage rate:

1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

2. By measuring the leakage rate of the system outside of the containment isolation valves at Pa, 34.7 psig, on the schedule required by Specification 4.6.1.2, and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

- The change from 400 to 500 °F represents plant specific data and is consistent with the WNP-2 FSAR (page 6.2-74, amen. #11).

- The preheating to 500°F is intended to preclude catalyst poisoning (as discussed in the FSAR) therefore the prudent practice of testing the catalyst on a 5 year cycle as recommended by the manufacturer is adopted. In addition, the principal fouling mechanism for a catalyst of this type is moisture buildup. The 6 month heat up to 500°F in 90 minutes test will maintain the catalyst dry, thus supporting the 5 year cycle. Also, the handling and emission of H_2 & O_2 gas within the confines of the Reactor Bldg every planned refueling outage may constitute a unnecessary hazard to plant and personnel.

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DRYWELL ~~(AND SUPPRESSION CHAMBER)~~ HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.2 Two independent drywell ~~(and suppression chamber)~~ hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell ~~(and/or suppression chamber)~~ hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each drywell ~~(and suppression chamber)~~ hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Starting the system from the control room, and

2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least _______ cfm.

Delete this spec. (over)

This LCO will be deleted for the following reasons;

1. FSAR section 6.2.5, Combustible Gas Control in Containment analysis, states that "natural turbulence from diffusion and convection caused by the elevated temperatures" ensure no local pocket with greater than 4% hydrogen and 5% oxygen can occur within the containment. The mixing capability is available, if necessary. A Battelle Northwest experiment was presented as a basis for concluding that the gaseous mix within the containment would be uniform. WNP-2 is to inert the containment to less than 3.5% oxygen by volume, therefore reducing the concern for hydrogen concentrations above 4%.

2. The recombiner system at WNP-2 was designed in accordance with Reg. Guide 1.7. The system takes suction from both the drywell (at an elevated point) and wetwell. Operation of the system provides circulation of the containment atmosphere as required by RG 1.7. The WNP-2 design is essentially identical to LaSalle Unit I. Communication with LaSalle personnel indicate that adequate mixing has not been an issue and the LCO is not present in their tech. spec's.

3. The Mark II containment design is essentially free of any closed compartments that would accumulate hydrogen. It would appear that the concern for hydrogen pocket formation is principally a PWR concern given the presence of hydrogen in the coolant under normal operating conditions and the number of compartments in a typical PWR containment.

4. The ability to mix was provided in the original design and is available to the operator, however, in light of inerting the containment and the lack of committment to operate the system in the FSAR analysis, this system should be elliminated from tech. specs. and from effecting plant operation.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3 The drywell and suppression chamber atmosphere oxygen concentration shall be less than ~~4%~~ 3.5% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than ~~15%~~ 25 of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than ~~15%~~ 25 of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

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*See Special Test Exception 3.10.5.

• Limit changed due to recommendation from A/E (Burns & Roe); i.e. plant specific.

• The 25% power limit is requested to provide the flexibility, within the capacity of the turbine bypass valves capacity, for entering drywell upto 25% power for inspection capability. The 25% limit would provide sufficient protection against SRV actuation with personnel inside containment as the bypass valves would provide that protection. Similar flexibility has been provided other operating plants (Quad Cities is known to have that flexibility).

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3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent standby service water (SW) system subsystems, with each subsystem comprised of:

a. One OPERABLE SW pump, and

b. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water to the associated safety related equipment,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5 and *, the subsystem(s) associated with systems and components required OPERABLE by Specifications 3.4.9.1, 3.4.9.2, {3.5.2}, 3.9.11.1, and 3.9.11.2.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one SW subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both SW subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN** within the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the SW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

c. In OPERATIONAL CONDITION 5 with the SW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

d. In All Operational Conditions ~~*~~, declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 As Applicable. ~~The provisions of Specification 3.0.3 are not applicable.~~

*When handling irradiated fuel in the secondary containment.

**Whenever both SW subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

1) CHANGE: ACTION STATEMENT d. WAS REVISED TO BE APPLICABLE FOR ALL OPERATING CONDITIONS AND TO REFERENCE SPECIFICATION 3.8.1.1 IN ADDITION TO 3.8.1.2

JUSTIFICATION: SPECIFICATIONS 3.8.1.1 AND 3.8.1.2 REQUIRE THE DIESEL GENERATORS TO BE OPERABLE IN ALL OPERATIONAL CONDITIONS NOT JUST IN OPERATIONAL CONDITION *.

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SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system subsystem(s) shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a service water actuation test signal.

c. BY VERIFYING EACH SUBSYSTEM IS OPERABLE PER THE REQUIREMENTS OF SPECIFICATION 4.0.5.

1.) CHANGE: ADDED SURVEILLANCE REQUIREMENT 4.7.1.1.c

JUSTIFICATION: THE SW SYSTEM IS AN ASME CODE SYSTEM WHICH IS INCLUDED IN OUR ASME SECTION XI VALVE AND PUMP PROGRAM.

HIGH PRESSURE CORE SPRAY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 The High Pressure Core Spray Service Water System shall be OPERABLE with the system comprised of:

a. One OPERABLE HPCS service water pump, and

b. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water to the HPCS diesel generator.

APPLICABILITY: When the diesel generator is required to be OPERABLE.

ACTION:

With the HPCS Service Water System INOPERABLE, declare the HPCS diesel generator inoperable and take the ACTION required by Specifications 3.8.1.1 and 3.8.1.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The HPCS Service Water System shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a service water actuation test signal.

c. By verifying the system is OPERABLE per the requirements of Specification 4.0.5.

This LCO added to reflect plant specific design.

CHANGE: ADDED THE ENTIRE SECTION 3/4.7.1.2

JUSTIFICATION: THE HIGH PRESSURE CORE SPRAY SERVICE WATER SYSTEM IS A WNP-2 PLANT SPECIFIC SYSTEM WHICH IS REQUIRED TO BE INCLUDED IN TECHNICAL SPECIFICATIONS BUT WAS NOT INCLUDED IN EARLIER SUBMITTALS.

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PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.~~X~~ 3 The {ultimate heat sink} shall be OPERABLE with:

a. A minimum water level at elevation ~~433'3" + 0", -13"~~ 432'-9" Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 77°F.

~~(c. (At Least) (two) OPERABLE cooling tower fans.)~~

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

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the SW system inoperable and take the ACTION required by Specification 3.7.1.1.

c. In Operational Condition *, declare the SW system inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The {ultimate heat sink} shall be determined OPERABLE at least once per~~X~~

~~a.~~ {24 hours by verifying the water temperature and water level to be within their limits.

~~(b. 31 days by starting each cooling tower fan from the control room and operating the fan for at least 15 minutes.)~~

~~(c. 18 months by verifying that each (service water) cooling tower fan starts automatically when the associated (service water) loop is initiated.~~

*When handling irradiated fuel in the secondary containment.

1. CHANGE: CHANGED MINIMUM WATER LEVEL IN ULTIMATE HEAT SINK TO 432'-9"

JUSTIFICATION: 432'-9" IS PLANT SPECIFIC AND HAS BEEN DETERMINED TO PROVIDE ADEQUATE VOLUME TO SUPPLY THE REQUIRED COOLING FOR THE 30 DAY POST ACCIDENT REQUIREMENTS.

2. CHANGE: DELETED 3.7.1.3.c

JUSTIFICATION: OUR PLANT DOES NOT USE COOLING TOWERS AS PART OF OUR ULTIMATE HEAT SINK.

3. CHANGE: DELETED 4.7.1.3.b & c

JUSTIFICATION: OUR PLANT DOES NOT USE COOLING TOWERS AS PART OF OUR ULTIMATE HEAT SINK.

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3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2. Two independent control room emergency filtration system trains shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with one control room emergency filtration train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

1. With one control room emergency filtration train inoperable, restore the inoperable train to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE train in the ~~(isolation)~~ PRESSURIZATION mode of operation.

2. With both control room emergency filtration trains inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency filtration system train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the train operates for at least 10 hours with the heaters OPERABLE.

*When irradiated fuel is being handled in the secondary containment.

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3.7.2.b.1 Pressurization mode allows make up air from remote intake(s). Recirculation (isolation) mode does not. Make up air is monitored for radiation and chlorine

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SURVEILLANCE REQUIREMENTS (Continued)

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

1. Verifying that the train satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05 ~~(*)~~% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, ~~and the~~ train flow rate is 1000 cfm ± 10%. when operating

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~(**)~~ 1.0%; and

3. Verifying a train flow rate of 1000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~(**)~~ 1.0%.

d. ~~c.~~ At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the train at a flow rate of 1000 cfm ± 10%.

2. Verifying that on a chlorine detection isolation mode actuation test signal, the train automatically switches to the ~~isolation~~ Recirculation mode of operation and the isolation valves close within 10 seconds.

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4.7.2.b.1

- Flow rate is verified in 4.7.2.b.3.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that on each of the below pressurization mode actuation test signals, the train automatically switches to the pressurization mode of operation and the control room is maintained at a positive pressure of 1/8 inch W.G. relative to the outside atmosphere during train operation at a flow rate less than or equal to 1000 cfm:

   a) Drywell pressure-high,

   b) Air intake-high radiation

   c) Reactor vessel water level-low, and

   d) Reactor Building exhaust plenum-high radiation.

4. Verifying that the heaters dissipate 5.0 ± 0.5 Kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than ~~(*)%~~ 0.05% in accordance with ANSI N510-1975 while operating the train at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than ~~(**)%~~ 1.0% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the train at a flow rate of 1000 cfm ± 10%.

~~*0.05% value applicable when a~~ SINGLE ~~HEPA filter or charcoal adsorber efficiency of 99% is assumed,~~ AND ~~or 1%~~ SINGLE ~~when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)~~

(**0.175% value applicable when a charcoal adsorber efficiency of 99% is assumed, or 1% value applicable when a charcoal adsorber efficiency of 95% is assumed, or 10% value applicable when a charcoal adsorber efficiency of 90% is assumed in the NRC staff's safety evaluation.)

THE FSAR values for filter efficiency are referenced in the WNP-2 Safety Evaluation Report and have been found acceptable by the NRC. The FSAR assumes 99% efficiency for HEPA filters and 95% for ~~carbon~~ charcoal adsorbers. Therefore leakage limits of 0.05% and 1.0% are appropriate for the HEPA and charcoal filters, respectively.

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3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. 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.

3. Verifying that the pump flow controller is in the correct position.

b. ~~(At least once per 92 days)~~ When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at ~~1000 + 20, - 80~~ ≥900 psig.*

INPUT FROM GE COMMENT

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor ~~steam~~ pressure is adequate to perform the test.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation {and restart} and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, -0 psig.*

~~3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.~~

delete. Logic System functional test included in 3/4.3.5 ∴ this requirement is redundant. Performed at same frequency.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

PLANT SYSTEMS

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers summarized in Table 3.7.4-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Examinations

The first inservice visual examination of snubbers shall be performed during the first refueling outage and shall include all snubbers listed in Table 3.7.4-1. Subsequent visual examinations shall be performed in accordance with the following schedule:

Number of Unexplained Inoperable Snubbers per Examination Period	Subsequent Visual Examination Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two major groups: those accessible and those inaccessible during reactor operation. Subgroups may be established based on snubber physical characteristics, service application, or environmental conditions. Each group or subgroup may be examined independently. Snubbers accessible during reactor operation shall be examined in accordance with the above schedule. Examination of snubbers inaccessible during reactor operation shall occur during each reactor shutdown greater than 48 hours unless examinations were previously performed in accordance with the above schedule.

An advance copy of this specification has been sent to Horace Shaw, NRC-MEB.

*The inspection interval shall not be lengthened more than one step at a time.
#The provisions of Specification 4.0.2 are not applicable.

4.7.4.a • WNP-2 programs for preservice inspection of snubbers entails extensive visual examination, including selected examinations at elevated temperatures. Hence the 4-10 month requirement is excessive. (Reference WNP-2 PSI Program Plan and Power Ascension Test program)

- Grouping of snubbers will facilitate sample selection and promote identification of trends

- The Sample selection options are designed to ensure, by funct test, that 90% -100% of the plants snubbers are operable. Use of approved maintenance procedures to install snubbers and severely restricted personnel access to plant areas will ensure continued snubber operability. Additionally, WNP-2 has an inert containment atmosphere. Approximately half of WNP-2's snubbers are inside containment. Rigid adherence to a visual ~~inspection~~ examination frequency would require extended shutdown and de-inerting time.

b. Visual Examination Acceptance Criteria

Visual examinations shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) that attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of these visual examinations may be determined OPERABLE for the purpose of establishing the next visual examination interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Surveillance Requirement 4.6.4.d.

c. Functional Tests

For the purpose of functional testing of safety related snubbers, the sample plan and resample plan should provide a confidence level of 95% that 90% to 100% of the snubbers are operable. Several sample plans are known to meet this criteria. These sample plans are listed as options to allow latitude for continued investigation leading to the implementation of the most effective sample plan.

The NRC Regional Administrator shall be notified in writing of the option selected for implementation prior to the initiation of the functional testing program. If notice is not given prior to the initiation of the functional testing program, that option implemented in the most recent refueling outage shall be implemented.

SAMPLE PLAN (OPTIONS)

OPTION #1

During the first refueling outage and at least once per 18 months thereafter during shutdown, a representative sample of at least that number of snubbers which follows the expression $35\ (1 + \frac{c}{2})$, where c = 3, the allowable number of snubbers not meeting the acceptance criteria, shall be functionally tested either in-place or in a bench test. For each number of snubbers above c which does not meet the functional test acceptance criteria of Specification 4.7.4.d, an additional sample selected according to the expression: $35\ (1 + \frac{c}{2})\ (\frac{2}{c + 1})^2\ (a - c)$ shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the representative sample.

Functional testing shall continue according to the expression $b\ [35\ (1 + \frac{c}{2})\ (\frac{2}{c + 1})^2]$ where b is the number of snubbers found inoperable in the previous re-sample, until no additional inoperable snubbers are found within a sample or until all snubbers in Table 3.7.4-1 have been functionally tested.

The option plans are designed to allow the utility flexibility in its snubber testing program. As noted, each option provides the same level of confidence in snubber serviceability.

OPTION #2

During each refueling outage, a representative sample of at least 37 snubbers listed in Table 3.7.4-1 shall be functionally tested. Additional testing shall be in accordance with Figure 4.7-1 which includes acceptance and rejection criteria. "C" is the cumulative total number of snubbers found not meeting the functional test acceptance criteria. The cumulative number of snubbers tested is denoted by "N". At the end of each testing day, the new values of "N" and "C" shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that group shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of that group of snubbers shall be terminated. When the point plotted falls in the "Continue Testing" region, additional snubbers shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that group have been tested.

OPTION #3

During each refueling outage, a representative sample of 55 snubbers listed in Table 3.7.4-1 shall be functionally tested. For each snubber failing the functional test acceptance criteria another sample of at least 1/2 the initial lot shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, "1 ÷ C/2" where "C" is the number of snubbers found failing the functional test acceptance criteria. Another sample of at least 1/2 of the initial test lot shall be tested for each subsequent snubber determined to fail the functional test acceptance criteria.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber, if it is repaired and installed in another position, and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For any snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. When tested at $\geq$ 10% of rated snubber load, the force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. Additionally, mechanical snubber drag force greater than 150% of the previously measured value shall be noted in the test record as an indication of impending failure. Impending failure shall not be counted as failed snubbers. Corrective action shall be documented per paragraph 4.7.4.e.

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

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

4. Testing methods which measure parameters indirectly or measure parameters other than those specified may be used provided, (1) the results can be correlated to the specified parameters through established methods and, (2) that activation testing is performed at $\geq$ 10% of the rated snubber capacity.

e. Snubber Service Records

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.

f. Exemption From Visual Examination or Functional Tests

Permanent or other exemptions from surveillance requirements for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented. Exempt snubbers shall be indicated in Table 3.7.4-1.

4.7.4.d.1. Increased drag force does not, per se, render a snubber inoperable. As long as maximum drag force is not exceeded, the snubber shall be considered acceptable.

4.7.4.d.4 This allows flexibility to use alternative testing techniques if and when such techniques are developed.

4.7.4.f. Various national codes (eg ASME) allow exemption from examination requirements based on industrial experience. This provision enables exemption based on plant specific experience or based on future standards (IWF) when the Commission is given adequate justification by the owner.

TABLE 3.7.4-1

All WNP-2 safety related mechanical snubbers are manufactured by Pacific Scientific (PSA).

Safety Related Mechanical Snubbers* +

System Snubbers Installed On	Snubber Size	Number of Snubbers
AS	½	1
	1	1
COND	1	2
DE	¼	3
	½	1
	1	10
	3	7
FPC	½	2
	1	3
	3	1
HPCS	3	8
	10	6
HY	¼	4
LPCS	3	4
	10	3
MS	¼	24
	½	15
	1	7
	3	7
	35	30
	100	8
MSLC	¼	1
	½	3
	1	1
MSRV	3	2
	10	129
	35	2

System Snubbers Installed On	Snubber Size	Number of Snubbers
RCC	¼	23
	½	13
	1	12
	3	4
RCIC	¼	2
	½	5
	1	19
	3	28
	10	5
RFW	1	2
	10	26
	35	3
	100	2
RHR	¼	14
	½	16
	1	44
	3	108
	10	51
	35	18
	100	1
RRC	¼	12
	½	5
	1	11
	35	35
	100	8
RWCU	¼	18
	½	20
	1	11
	3	14
	10	1
	100	2

System Snubbers Installed On	Snubber Size	Number of Snubbers
SGT	3	2
	10	2
SLC	¼	5
	½	5
	1	3
SW	3	1
	10	8
	35	2
VR	½	5
	1	2
TOTAL		853

*Snubbers may be added to safety related systems without prior License amendment to Table 3.7.4-1 provided that a revision to Table 3.7.4-1 is included with the next License Amendment Request.

+This Table presents a summary of plant safety-related snubbers. These snubbers are specifically identified in the WNP-2 Inservice Inspection Program component support listing which contains snubber part number

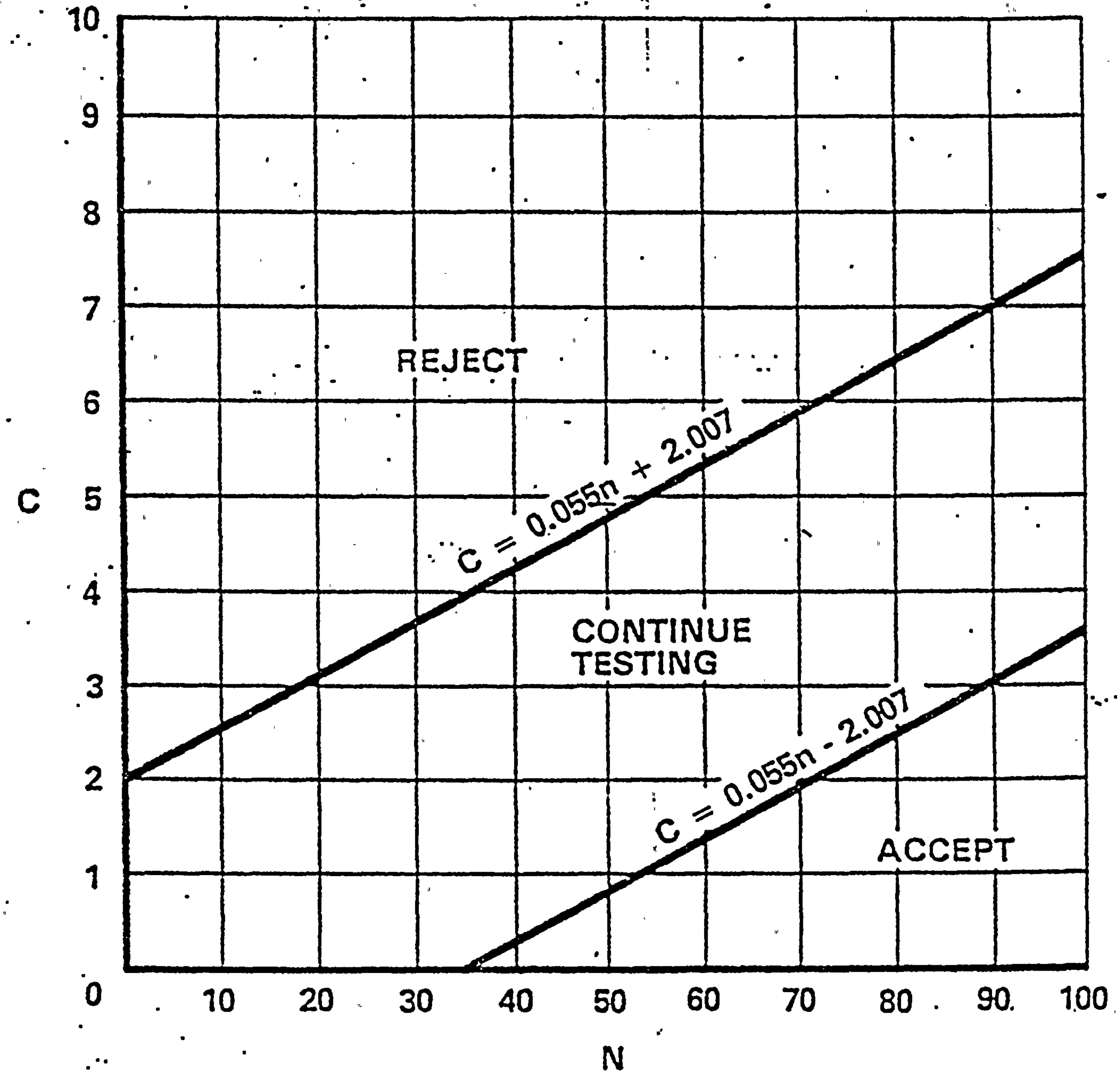

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.5.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.5.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

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SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.5.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

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3/4.7.6 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The fire suppression water system shall be OPERABLE with:

a. At least ~~three~~ two OPERABLE fire suppression pumps, ~~each~~ with a capacity of 2000 gpm, with their discharge aligned to the fire suppression header. pumping from the circulating water basin or one 2500 gpm diesel driven pump supplied from the secondary water supply tank,

b. Two separate fire water supplies, with a minimum contained volume of ~~370,000~~ 300,000 gallons in the recirculating water pump house inlet basin and 280,000 gallons in the ~~ground level storage~~ secondary water supply tank.

c. An OPERABLE flow path capable of taking suction from the circulating water pump house inlet basin and the ~~ground level storage~~ secondary water supply tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.6.2, 3.7.6.4, and 3.7.6.5.

APPLICABILITY: At all times.

ACTION:

a. With ~~one~~ two 2000 gpm or the 2500 gpm of the above required fire pumps and/or one water supply inoperable, restore at least ~~three~~ two 2000 gpm & one 2500 gpm fire pumps and two fire water supplies to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the minimum contained water supply volume.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven fire suppression pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

* The WNP-2 fire suppression system consists of 2 electric and 1 diesel driven 2000 gpm capacity pumps that are supplied by the circulating water pump basin. The backup is the 2500 gpm diesel driven unit that is supplied from the secondary water supply tank. The capacity of either one 2500 gpm unit or two 2000 gpm units provides sufficient fire suppression. The changes made are intended to reflect the WNP-2 design.

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SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 6 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that ~~each~~ the circulating water basin supplied fire suppression pumps develop~~s~~ at least 2000 gpm at a system head of 250 feet and that the secondary water supplied unit develops 2500 gpm at a system head of 325 feet.

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 95 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.6.1.2 ~~Each~~ Both diesel driven fire suppression pumps shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying the fuel storage tanks contain~~s~~ at least 150 gallons of fuel.

2. Starting the diesel driven pump from ambient conditions and operating for greater than or equal to 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.6.1.3 Each diesel driven fire pump starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each ~~(pilot)~~ cell is above the plates,

2. The ~~(pilot)~~ cell specific gravity, corrected to (77)°F and full electrolyte level, is greater than or equal to (1.200),

3. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. Battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

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SPRAY AND SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.2 The following pre-action and deluge spray and sprinkler systems shall be OPERABLE:

a. Radwaste Building:

1. Cable spreading room, elev. 484', system #65.
2. Cable chase and corridor, elev. 441' to 525', system #66.
3. Control Bldg. emergency charcoal filters, elev. 525', system ~~#~~ #WMA-DV-54A and WMA-DV-54B

b. Diesel Generator Building:

1. DG room 1A and day tank room, elev. 441', system #79.
2. DG 1B day tank pump room, elev. 441', system #80.
3. DG room 1B and day tank room, elev. 441', system #81.
4. DG 1B day tank pump room, elev. 441', system #82.
5. HPCS DG room and day tank room, elev. 441', system #83.
6. HPCS DG day tank pump room, elev. 441', system #84.

c. Reactor Building:

1. Standby gas treatment system charcoal filters, elev. 572', system ~~#~~ (over)

APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each of the above required spray and sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

Deluge ~~system~~

SGT-DIV-1A-1
SGT-DIV-1A-2
SGT-DIV-1A-3
SGT-DIV-1B-1
SGT-DIV-1B-2
SGT-DIV-1B-3

2. Sump vent filter system charcoal filters, elev. 572',
system # REA-DV-2A and 2B.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a detector test signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each {deluge} nozzle's spray area to verify that the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray and sprinkler header and verifying each open head spray and sprinkler nozzle is unobstructed.

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HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.3 The eighteen Halon systems in the PGCC units in the control room shall be OPERABLE with the storage tanks having at least 95% of full charge ~~(level)~~ weight and 90% of full charge pressure.

APPLICABILITY: Whenever equipment protected by the Halon systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight and pressure ~~(level)~~.

c. At least once per 18 months by:

1. Verifying the system, including associated ventilation system fire dampers and fire door release mechanisms, actuates, manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through (accessible)* headers and nozzles to assure no blockage.

(*Accessible headers and nozzles.)

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.6.4 The fire hose stations shown in Table 3.7.6.4-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7.6.4-1 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.4 Each of the fire hose stations shown in Table 3.7.6.4-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

1. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the station.

2. Removing the hose for inspection and re-racking, and

3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

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TABLE 3.7.6.4-1

FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK IDENTIFICATION
1.	Reactor Bldg. Standpipe RB-1	427'-0"	RB-HS-11
2.	Reactor Bldg. Standpipe RB-1	446'-0"	RB-HS-12
3.	Reactor Bldg. Standpipe RB-1	476'-0"	RB-HS-13
4.	Reactor Bldg. Standpipe RB-1	506'-0"	RB-HS-14
5.	Reactor Bldg. Standpipe RB-1	527'-0"	RB-HS-15
6.	Reactor Bldg. Standpipe RB-1	553'-0"	RB-HS-16
7.	Reactor Bldg. Standpipe RB-1	577'-0"	RB-HS-17
8.	Reactor Bldg. Standpipe RB-1	612'-0"	RB-HS-18
9.	Reactor Bldg. Standpipe RB-2	427'-0"	RB-HS-21
10.	Reactor Bldg. Standpipe RB-2	446'-0"	RB-HS-22
11.	Reactor Bldg. Standpipe RB-2	476'-0"	RB-HS-23
12.	Reactor Bldg. Standpipe RB-2	506'-0"	RB-HS-24
13.	Reactor Bldg. Standpipe RB-2	527'-0"	RB-HS-25
14.	Reactor Bldg. Standpipe RB-2	553'-0"	RB-HS-26
15.	Reactor Bldg. Standpipe RB-2	577'-0"	RB-HS-27
16.	Reactor Bldg. Standpipe RB-2	612'-0"	RB-HS-28
17.	Railroad Car Air Lock	446'-0"	RB-HS-29
delete ~~18.~~	~~Diesel Generator Bldg.~~	~~446'-0"~~	~~DG-HS-40~~
18. ~~19.~~	Radwaste Bldg. Standpipe RWB-1	472'-0"	RWB-HS-13
19. ~~20.~~	Radwaste Bldg. Standpipe RWB-1	492'-0"	RWB-HS-14
20. ~~21.~~	Radwaste Bldg. Standpipe RWB-1	512'-0"	RWB-HS-15
21. ~~22.~~	Radwaste Bldg. Standpipe RWB-1	530'-0"	RWB-HS-16
22. ~~23.~~	Turbine Generator DG Bldg. Corridor	446'-0"	RWB-HS-25
23. ~~24.~~	Radwaste Bldg. Stair A-13	472'-0"	RWB-HS-26
24.	Radwaste Bldg. In corridor	492'-0"	RWB-HS-28
25.	Radwaste Bldg. In corridor	472'-0"	RWB-HS-29
26.	Radwaste control room corridor	506'-0"	RWB-HS-31
27.	Diesel Generator Bldg.	446'-0"	DG-HS-40
28.	Diesel Generator Bldg. In Corridor	446'-0"	DG-HS-41

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YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.6.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7.6.5-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7.6.5-1 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7.6.5-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months, during March, April or May and during September, October or November, by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

   2. Replacement of all degraded gaskets in couplings.

   3. Performing a flow check of each hydrant.

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TABLE 3.7.6.5-1

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION		HYDRANT NUMBER
1.	South side of Diesel Generator Bldg.	HT-1A
2.	Southeast corner of Diesel Generator Bldg.	HT-1B
3.	West side of Radwaste Bldg.	HT-1G
4.	South side of Radwaste Bldg.	HT-1H
5.	Northwest of Standby Service Water Pump House 1A	HT-1M
6.	North side of Standby Service Water Pump House 1B	HT-1N
7.	West side of Radwaste and Turbine Generator Bldg.	HT-1R

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3/4.7.7 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.7 All fire barrier assemblies, including walls, floor/ceilings, cable tray enclosures and other fire barriers, separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area, and all sealing devices in fire rated assembly penetrations, including fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals and ventilations seals, shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour establish a continuous fire watch on at least one side of the affected assembly(s) and/or sealing device(s) or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly(s) and/or sealing device(s) and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each of the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE at least once per 18 months by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly.

b. Each fire window, fire damper and associated hardware.

c. At least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrical supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. That each locked-closed fire door is closed at least once per 7 days.

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test of these mechanisms at least once per 18 months.

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

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PLANT SYSTEMS

3/4.7.8 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The temperature of each area shown in Table 3.7.8-1 shall be maintained within the limits indicated in Table 3.7.8-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.8-1:

a. For more than eight hours, in lieu of any report required by Specification 6...1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.8 The temperature in each of the areas shown in Table 3.7.8-1 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
a. Control Room	< 104*
b. Auxiliary Electric Equip. Rooms	< 104*
~~c. Diesel Generator Rooms~~ Delete, not subject to accident environment at WNP-2	~~< 135°~~
d. HPCS, LPCS, RHR, RCIC Rooms	< 150*
e. Primary Containment Beneath Reactor Pressure Vessel	< 165*
f. Switchgear Rooms	< 104*
c. Primary Containment (Drywell)	< 150

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~~3/4.7.9~~ MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 The main turbine bypass system shall be OPERABLE.

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

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or ~~take the ACTION required by Specification 3.2.3~~ reduce Thermal POWER to less than 25% rated thermal power within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.9 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME ~~to be less than or equal to 100 milliseconds to a valve position equivalent to 80% of rated bypass flow~~ meets the following requirements when measured from the initial movement of main turbine throttle valves or control valves:

   a. 80% of turbine bypass system capacity shall be established within 0.3 seconds.

   b. Bypass valve opening shall start within .1 seconds.

ACTION

WNP-2 analysis does not presently identify the MCPR penalty associated with the absence of bypass system.

4.7.9 Changes recommended by GE.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

1. Separate day ~~and engine mounted fuel~~ tanks containing a minimum of 1400 gallons of fuel,

2. A ~~separate~~ fuel storage system containing a minimum of 53,000 ~~54,000~~ gallons of fuel for DG-1 and DG-2, and 30,000 gallons of fuel for DG-3.

3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator 1 or 2 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours 24 thereafter; restore at least two offsite circuits and diesel generators 1 and 2 to OPERABLE status within 72 hours 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one offsite circuit and diesel generator 1 or 2 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators 1 and 2 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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3.8.1.1.b.2 53,000 GALLONS. PLANT SPECIFIC

B&R DESIGN CALCULATIONS 5.43.02 USING ACTUAL FUEL CONSUMPTION TEST RESULTS PROVIDED BY THE FACTORY SHOWS A CONSUMPTION RATE OF 5.4 gallons per minute at full rated load. THESE ARE PROVIDED.

3.8.1.1. ACTION a CHANGE. 8 hours to 24 hours, 72 hours to 7 days.

BASIS: 1) OTHER BWRS TEST REDUNDANT DIESEL GENERATOR ONCE PER 24 HOURS FOR UP TO 7 DAYS.

2) CONDITIONS WHICH COULD RENDER THE DIESEL GENERATOR INOPERABLE ARE INSTRUMENTED AND ALARMED.

3) EXCESSIVE HIGH SPEED STARTS WITHOUT SUBSEQUENT LOADING TO AT LEAST 50% OF RATED LOAD MAY BE DETRIMENTAL TO THE RELIABILITY OF THE UNIT. LOADING OF THE DIESEL GENERATOR UNIT IN THIS PLANT STATUS IS NOT RECOMMENDED.

3.8.1.1.b.1 PLANT SPECIFIC

DRAFT

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With diesel generator 3 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore the inoperable diesel generator 3 to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.
24 hours 7days

d. With diesel generator 1, 2, or 3 of the above required A.C. electrical power sources inoperable, in addition to ACTION a, b or c, as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 for one diesel generator at a time, within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

f. With diesel generators 1 and 2 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators 1 and 2 to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators 1 and 2 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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~~ACTION e) 72 hours to 7 days.~~

~~BASIS: TO MAKE THIS ACTION CONSISTENT WITH ACTION ITEM a.~~

ACTION ~~X~~ c) 6 HOURS TO 24 HOURS, 72 HOURS TO 7 days.

BASIS: SAME AS ACTION a CHANGE BASIS.

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day ~~and engine-mounted fuel~~ tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump MANUALLY starts and transfers fuel from the storage system to the day ~~and engine-mounted fuel~~ tank~~s~~.

4. ~~Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds.~~ 60 HZ FOR 1 & 2 & 13 SEC. FOR 3 The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± ~~3.0~~ Hz within ~~(13)~~ 10 seconds FOR 1 & 2 AND 13 SEC FOR 3 after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.
   b) Simulated loss of offsite power by itself.
   c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   d) An ESF actuation test signal by itself.

5. Verifying the diesel generator is synchronized, loaded to greater than or equal to 4400 kw for diesel generators 1 and 2 and 2600 kw for diesel generator 3 in less than or equal to 60 seconds, and operates with these loads for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 230 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day ~~and engine-mounted fuel~~ tanks.

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4.8.1.1.2.a.1 PLANT SPECIFIC

4.8.1.1.2.a.3 ADD WORD, MANUALLY

BASIS: IN ORDER FOR THE [illegible] TO START AUTOMATICALLY THE DIESEL GENERATOR WOULD HAVE TO RUN LOADED FOR APPROXIAMATELY FOUR HOURS TO LOWER THE FUEL LEVEL IN THE DAY TANK TO A POINT WHERE THE LEVEL SWITCH WOULD START THE PUMP.

DELETE "AND ENGINE [illegible] FUEL". BASIS PLANT SPECIFIC.

4.8.1.1.2.a.4 DELETE 900 RPM & INSERT 60 HZ - PLANT SPECIFIC DG1 AND DG2 DO NOT HAVE RPM INDICATION. FREQUENCY IS DIRECTLY CORRELATED TO ENGINE SPEED.

CHANGE - 12 SECONDS TO 10 SECONDS

BASIS: REGULATORY GUIDE 1.108 SPECIFIES 10 SECONDS.

4.8.1.1.2.b PLANT SPECIFIC

c. At least once per ~~(31) [if ground water table is equal to or higher than the bottom of the tank])~~ (92) days by removing accumulated water the fuel storage tank(s).

d. At least once per 92 days and from new fuel oil prior to addition to the storage tanks, by obtaining a sample obtained in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

~~1.~~ As soon as sample is taken or from new fuel prior to addition to the storage tank, as applicable, verify in accordance with the tests specified in ASTM-D975-77 that the sample has:

a) A water and sediment content of less than or equal to 0.05 volume precent.

b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

c) A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to .8299 but less than or equal to .9762 or an API gravity @ 60°F of greater than or equal to 30 degrees.

~~2. Within one week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.~~

e. ~~3.~~ Within two weeks after obtaining the sample from new fuel oil after addition, verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137, Position 2.a, are met when tested in accordance with ASTM-D975-77.

f. ~~e.~~ At least once per 18 months, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

Plant Specific

2. Verifying the diesel generator capability to reject a load of greater than or equal to ~~1258~~ 1377 kw for diesel generator 1, greater than or equal to ~~1258~~ kw for diesel generator 2, and greater than or equal to ~~2228~~ 2380 kw for diesel generator 3 while maintaining engine speed ≤ 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 4400 kw for diesel generators 1 and 2 and 2600 kw for diesel generator 3 without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

4.8.1.1.2.c Plant Specific - Water table is well below (approximately 60 feet) level of tanks.

4.8.1.1.2.d.2 ASTM D2274-74, "Oxygen Stability of Distillate Fuel Oil (Accelerated Method)", measures the stability of distillate fuels under accelerated oxidizing conditions. The standard itself points out the tentative nature of this test and that any correlations between this test and field conditions may vary considerably. In other words, it is not indicative of the actual stability of the oil in the underground tanks. It is a time consuming, costly test which provides results which are next to useless. Note 3 under Precision of the Standard states:

"The repeatability and reproducability of this method are still under study."

f7

4.8.1.1.2.e.3 Take the last sentence of Section 3 and put it at the end of Section 2. The requirement is necessary to protect other loads on the diesel generator. It is not applicable on Section 3.

4.8.1.1.2.e (was d.3)

The testing requested verifies the grade of diesel fuel and impurity levels on a 92 day frequency. The testing conducted in d.1 is sufficient to indicate fuel degradation. Performing the ASTM-D975-77 and RG 1.137 testing on a frequency consistent with fuel receipt is considered adequate. Neither R.G. 1.137 nor referenced ANSI standard requires ASTM-D975-77 testing on a 92 day frequency.

SURVEILLANCE REQUIREMENTS (Continued)

4. Simulating a loss of offsite power by itself, and:

   a) For divisions 1 and 2:

      1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the autoconnected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 3.0 Hz during this test.

   b) For division 3:

      1) Verifying de-energization of the emergency bus. 13

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with ~~the~~ permanently connected ~~(1ss)~~ loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 3.0 Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 3.0 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) For divisions 1 and 2:

      1) Verifying deenergization of the emergency busses and loads shedding from the emergency busses.

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected ~~(shutdown)~~ loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± *3.0* Hz during this test.

b) For division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads and the auto-connected emergency loads within *30* seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± *3.0* Hz during this test.

8. Verifying that all automatic diesel generator trips are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal except:

a) For division 1 and 2, engine overspeed and generator differential current, INCOMPLETE STARTING SEQUENCE, EMERGENCY MANUAL STOP.

b) For division 3, engine overspeed, generator differential current and emergency manual stop.

4650

9. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4900 kw for diesel generators 1 and 2 and 2850 kw for diesel generator 3. During the remaining 22 hours of this test, the diesel generator shall be loaded to 4400 kw for diesel generators 1 and 2 and 2600 kw for diesel generator 3. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± *3.0* Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this

FOR 1 & 2 AND BSC. FOR 3

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5.a. COMPLETE THE LIST.

9. 2000 HR RATING OF DIESEL

SURVEILLANCE REQUIREMENTS (Continued

test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.e.4.a)2) and b)2).*

10. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4650 kw for diesel generators 1 and 2 and 2600 kw for diesel generator 3.

11. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

12. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

13. Verifying ~~that with all diesel generator air start receivers pressurized to less than or equal to 215 psig and the compressors secured, the diesel generator starts at least~~ 5 times ~~from ambient conditions and accelerates to 900 rpm ± (5)% in less than or equal to 10 seconds~~.

~~14. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day and engine-mounted tanks of each~~ diesel ~~via the installed cross-connection lines.~~

15. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 1 and 2.

16. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) ~~(Turning gear engaged.)~~ MAINTENANCE LOCKOUT KEY-LOCK SWITCH
    b) (Emergency stop.)

*If Surveillance Requirement 4.8.1.1.2.e.4.a)2) and/or b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at <continuous rating> kw for one hour or until operating temperature has stabilized.

4.8.1.1.2.e.13 This paragraph requires demonstration of five starts of the generators utilizing the amount of air retained in the air receivers with pressure not more than 215 psig. The design criteria for WNP-2 is five starts with the air receivers initially filled to 250 psig. This criteria has or will be proven during component testing at WNP-2. The number of starts the diesel generators are capable of making on any given amount of air is contingent on three factors:

1. Capacity of the air receivers,
2. The condition of the air start motors, and
3. The condition of the engines.

The capacity of the air receivers is not subject to change and, therefore, should not be the subject of periodic testing. The condition of the air start motors and the engines is subject to change; however, these changes tend to be very insidious and would be more readily detectable by maintaining a record of start/load times and load carrying capability, which is done every 31 days per Tech. Spec. 4.8.1.1.2.a.4, and every 18 months per Tech. Spec 4.8.1.1.2.e.9. There is no requirement for this test in Reg. Guides 1.108 and 1.9, IEEE Std. 387-1977, nor is it recommended by NUREG/CR.0660 (Enhancement of On-Site Emergency Diesel Generator Reliability).

4.8.1.1.2.e.14 This paragraph requires verification that the fuel transfer pump transfers fuel from each storage tank to each day tank via installed cross connect lines. Regulatory Position C.2.a(7) (Reg. Guide 1.108) only requires this demonstration if switching is required to satisfy the seven day storage requirement. The physical configuration of the fuel storage and transfer systems provides for a separate seven day supply of fuel for each of the three divisions without switching or transferring between divisions. (See WNP-2 FSAR, Chapter 9, Paragraph 9.5.4.3.)

4.8.1.1.2.e.16 Plant Specific

SURVEILLANCE REQUIREMENTS (Continued)

f. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that all three diesel generators accelerate to ~~at least 900 rpm~~ >60 HZ in less than or equal to ~~(~~10~~)~~ seconds.

g. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank ~~using a (sodium hypochlorite) solution,~~ MANUALLY AND IF NECESSARY WITH A SOLUTION OF TRISODIUM PHOSPHATE and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section ~~II Article IWD-5000.~~ XI, DIVISION 1 ART. IWD-5000

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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f) DG1 AND DG2 DO NOT HAVE RPM INDICATION.

g) IDENTIFY CORRECT ARTICLE

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. ~~For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests."~~ Entry into this test schedule shall be made at the 31 day test frequency.

BASIS: REGULATORY GUIDE 1.108 SECTION C.2.D STATE.

"AFTER COMPLETION OF THE DIESEL GENERATOR UNIT RELIABILITY DEMONSTRATION UNDER REGULATORY POSITION C.2.A.(9), THE INTERVAL FOR PERIODIC TESTING UNDER REGULATORY POSITION C.2.C... SHOULD BE NO MORE THAN 31 DAYS AND SHOULD DEPEND ON DEMONSTRATED PERFORMANCE..."

THIS IS INTERPRETED TO MEAN THAT OUR TESTING STARTS AT THIS TIME AND THIS TESTING HISTORY SHOULD NOT BE RENDERED INAPPLICABLE.

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FROM THE APPLICANT

DRAFT

ELECTRICAL POWER SYSTEMS

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator 1 or 2, and diesel generator 3 when the HPCS system is required to be OPERABLE, with each diesel generator having:

1. Day and engine mounted fuel tanks containing a minimum of 1400 gallons of fuel. see AC source operating

2. A fuel storage system containing a minimum of ~~54,000~~ 53,000 gallons of fuel for DG-1 and DG-2, and 30,000 gallons of fuel for DG-3.

3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the offsite circuits and/or diesel generators 1 or 2 of the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, in OPERATIONAL CONDITION 5, with the water level less than (22) feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator 3 of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

*When handling irradiated fuel in the secondary containment.

LIMITING CONDITION FOR OPERATION (Continued

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APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. For A.C. power distribution:

1. With less than Division 1 and ~~and/or~~ Division 2 of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

2. With Division 3 of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

b. For D.C. power distribution:

1. With less than Division 1 and/or Division 2 of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the Auxiliary Building and Enclosure Building and operations with a potential for draining the reactor vessel.

2. With Division 3 of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the ~~busses/MCCs/panels.~~ medium voltage buses.

*When handling irradiated fuel in the secondary containment.

action a.1 change justification :

Inclusion of "and" is consistent with LaSalle tech. spec. and division 3 is available as a source of water.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be de-energized*:

a. Circuits supplied by breakers 2AR and 2BR, MCC E-MC-8C.

b. Circuits supplied by panel E-LP-6BAG.

c. Circuits supplied by panel E-LP-3DAG.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be de-energized at least once per 24 hours** by verifying that the associated circuit breakers are in the tripped condition.

*Except during entry into the drywell.
**Except at least once per 31 days if locked, sealed or otherwise secured in the tripped condition.

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PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 6.9 kV circuit breakers, de-energize the 6.9 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by removing the fuses within 72 hours and verify the fuses associated with the inoperable breaker(s) to be removed at least once per 7 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 6.9 kV circuits which have their redundant circuit breakers tripped or to 480 volt circuits which have the fuses associated with the inoperable circuit breaker removed.

SURVEILLANCE REQUIREMENTS

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the medium voltage, 6.9 kV, circuit breakers are OPERABLE by ~~selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level and~~ performing:

      a) A CHANNEL CALIBRATION of the associated protective relays, and

      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

      ~~c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.~~

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing ~~a representative sample of at least 10% of~~ each ~~type of~~ lower voltage circuit breaker~~s~~. ~~Circuit breakers selected for functional testing shall be selected on a rotating basis.~~ Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the longtime delay trip element and 150% of the pickup of the short time delay trip element, and verifying that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. ~~For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.~~

SEE INSERT.

3. ~~By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.~~

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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TABLE 3.8.4.2-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER	SYSTEMS POWERED
a. 6.9 KV Circuit Breakers	
E-CB-RRA	RCC-P-1A
E-CB-RRA	RCC-P-1A
E-CB-RRB	RRC-P-1B
E-CB-RRB	RRC-P-1B
b. 480 VAC Fused Disconnects	
RRC-DISC-7C4A	RRC-V-67A
RCC-DISC-7C2A	RCC-V-71A
RCC-DISC-7C2B	RCC-V-72A
RCC-DISC-7C2C	RCC-V-17A
MS-DISC-8BA6D	MS-V-16
RWCU-DISC-8BA8B	RWCU-V-1
RHR-DISC-8BA2A	RHR-V-9
RCIC-DISC-8BA9D	RCIC-V-63
RCC-DISC-8BA10D	RCC-V-40
RHR-DISC-8BA6C	RHR-V-123B
RCIC-DISC-8BA6B	RCIC-V-76
CRA-DISC-8B4D	CRA-FN-1C2
CRA-DISC-8B3E	CRA-FN-1B1
CRA-DISC-8B5D	CRA-FN-1B2
CRA-DISC-8B6C	CRA-FN-2B1
CRA-DISC-8B5A	CRA-FN-2B2
CRA-DISC-8B3C	CRA-FN-5D
CRA-DISC-8B5B	CRA-FN-3B
CRA-DISC-8B5C	CRA-FN-3C
CRA-DISC-8B6A	CRA-FN-4B
CRA-DISC-8B3AL	CRA-AD-1B1
	CRA-AD-1B2
	CRA-AD-1C1
	CRA-AD-1C2
CRA-DISC-8B7C	CRA-AD-2B
RRC-DISC-8C5D	RRC-V-67B
E-DISC-8C2AR	PWR. RECEPTACLE
E-DISC-8C8AR	PWR. RECEPTACLE

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER	SYSTEMS POWERED
4.80 VAC Fused Disconnects (Continued)	
RC-DISC-8C5A	RRC-V-23B
RWCU-DISC-8C6A	RWCU-V-102
RWCU-DISC-8C6B	RWCU-V-106
RRC-DISC-8C8D	RRC-V-23A
RWCU-DISC-8C6D	RWCU-V-101
RWCU-DISC-8C6C	RWCU-V-100
RCC-DISC-8C7C	RCC-V-17B
RCC-DISC-8C7B	RCC-V-71C
RCC-DISC-8C7A	RCC-V-71B
CRA-DISC-7B2D	CRA-FN-1A2
CRA-DISC-7B4E	CRA-FN-1A1
CRA-DISC-7B5B	CRA-FN-2A2
CRA-DISC-7B5C	CRA-FN-2A1
CRA-DISC-7B2A	CRA-FN-5A
CRA-DISC-7B2B	CRA-FN-4A
CRA-DISC-7B1C	CRA-FN-5C
CRA-DISC-7B5A	CRA-FN-3A
CRA-DISC-7B6AR	CRA-AD-1A1, 1A2
CRA-DISC-7B4C	CRA-AD-2A
RCC-DISC-8C7D	RCC-V-72B
MS-DISC-8CB8A	MS-V-1
MS-DISC-8CB8B	MS-V-2
MS-DISC-8CB8C	MS-V-5
RHR-DISC-8BA5A	RHR-V-123A
MT-DISC-3DA2BL	MT-HOI-11
MT-DISC-3DA1D	MT-HOI-16
MT-DISC-3DA2CR	MT-HOI-19
c. 120 VAC Molded Case Circuit Breakers	
E-CB-PP7CAA/1	B35-C001A HEATER
E-CB-PP8CAA/8	B35-C001B HEATER

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4.8.4.2.2.1 THERE ARE ONLY FOUR 6.9. KV CIRCUIT BREAKERS PROTECTIVE RELAY MAINTENANCE FOR TYPE COM RELAYS, THE TYPE USED IN THESE BREAKERS, HAVE BEEN PREVIOUSLY SCHEDULED ONCE PER YEAR.

4.8.4.2.2.2 THERE ARE ONLY TWO LOW VOLTAGE CIRCUIT BREAKERS.

55 FUSED DISCONNECTS × 3 FUSE/DISC = 165 FUSES

10% × 165 = 16.5 ⇒ 17 ⇒ 18 ROUNDING TO MULT OF 3

REPLACING ~~18 FUSES/OUTAGE~~
×2
36 FUSES/OUTAGE (INCLUDES REDUNDANT BACKUP)

4.8.4.2.2.3 THE MANUFACTURER'S DATA REFERRED TO IN THIS SECTION IS FUSE RESISTANCE. IT IS NOT NORMALLY SUPPLIED DATA AND IS DIFFICULT TO OBTAIN

FUSE RESISTANCES USUALLY FALL INTO A BAND OF SMALL VALUES AND DO NOT DECREASE IN RESISTANCE AS TIME PASSES. ANY DETERIORATION OR PARTIAL BLOWS RESULT IN INCREASED RESISTANCE WHICH INCREASES THE POSSIBILITY OF CLEARING. THUS ANY DEGRADATION OF THE FUSE RESULTS IN INCREASED PROTECTION OF THE CIRCUIT.

VERBAL COMMUNICATION WITH FUSE MANUFACTURERS HAVE SUPPORTED THIS POSITION.

THE PURPOSE OF TESTING THE FUSES IN THIS SECTION IS TO VERIFY THEIR CAPABILITY TO PROVIDE PROTECTION FOR CONTAINMENT PENETRATION ASSEMBLIES.

THE TESTING RECOMMENDED BY THE GENERIC TECH SPEC DOES NOT PROVIDE THE FUNCTION THAT WAS INTENDED.

WNP-2 HAS UNDERGONE SUBSTANTIAL REDESIGN AND BACKFIT TO PROVIDE BOTH PRIMARY PROTECTION AND BACKUP PROTECTION FOR CONTAINMENT PENETRATION ASSEMBLIES.

MANUFACTURER'S TEST DATA, TIME CURRENT CLEARING CURVES, WERE PROVIDED TO NRR, PSB TO DEMONSTRATE ADEQUATE PROTECTION FOR PENETRATION ASSEMBLIES.
THIS IS THE APPLICABLE DATA TO DEMONSTRATE ADEQUATE PROTECTION.

WNP-2 DOES NOT CONSIDER ~~FUSE~~ RESISTANCE TESTING AN ADEQUATE MEANS TO VERIFY OPERABILITY OF THESE OVERCURRENT PROTECTIVE DEVICES.

WNP-2 DOES RECOGNIZE THE IMPORTANT FUNCTIONS PROVIDED BY THESE CIRCUITS.

WNP-2 PROPOSES TO CHANGE OUT ON A ROTATING BASIS 10% OF THE FUSES IDENTIFIED ON A 18 MONTH BASIS.

TECH. SPEC. WORDING.

INSERT:

3. BY SELECTING AND REPLACING A REPRESENTATIVE SAMPLE OF EACH TYPE OF FUSE ON A ROTATING BASIS. EACH REPRESENTATIVE SAMPLE OF FUSES SHALL INCLUDE AT LEAST 10% OF ALL FUSES OF THAT TYPE.

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each valve shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

SEE BACK FOR BASIS.

With the thermal overload protection for one or more of the above required valves inoperable, ~~(continuously)~~ bypass the inoperable thermal overload within 8 hours~~(, restore the inoperable thermal overload to OPERABLE status within 30 days)~~ or declare the affected valve~~(~~s~~)~~ inoperable and apply the appropriate ACTION statement~~(~~s~~)~~ for the affected system~~(~~s~~)~~.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for tne above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

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BASIS: THE CONCERN OF THE COMMISSION IS TO TEST THE MOV THERMAL OVERLOADS TO VERIFY THAT THEIR PROTECTIVE ACTION WILL NOT PREVENT THE COMPLETION OF THE VALVE FUNCTION. TO DECLARE THE VALVE INOPERABLE DUE TO THE FACT THE THERMAL OVERLOAD IS INOPERABLE WHEN BYPASSED IS INCONSISTENT WITH THE CONCERN OF THE COMMISSION.

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED	VALVE NUMBER	SYSTEM(S) AFFECTED
CAC-V-2		MSLC-V-1A	
CAC-V-4		MSLC-V-1B	
CAC-V-6		MSLC-V-1C	
CAC-V-8		MSLC-V-1D	
CAC-V-11		MSLC-V-2A	
CAC-V-13		MSLC-V-2B	
CAC-V-15		MSLC-V-2C	
CAC-V-17		MSLC-V-2D	
		MSLC-V-3A	
CIA-V-20		MSLC-V-3B	
CIA-V-30A		MSLC-V-3C	
CIA-V-30B		MSLC-V-3D	
		MSLC-V-4	
FPC-V-153		MSLC-V-5	
FPC-V-154		MSLC-V-9	
FPC-V-156		MSLC-V-10	
HPCS-V-1		RCC-V-5	
HPCS-V-4		RCC-V-6	
HPCS-V-10		RCC-V-17A	
HPCS-V-11		RCC-V-17B	
HPCS-V-12		RCC-V-21	
HPCS-V-15		RCC-V-40	
HPCS-V-23		RCC-V-71A	
		RCC-V-71B	
LPCS-V-1		RCC-V-71C	
LPCS-V-5		RCC-V-72A	
LPCS-FCV-11		RCC-V-72B	
LPCS-V-12		RCC-V-104	
		RCC-V-129	
MS-V-1		RCC-V-130	
MS-V-2		RCC-V-131	
MS-V-5			
MS-V-16		RCIC-V-1	
MS-V-19		RCIC-V-8	
MS-V-20		RCIC-V-10	
MS-V-67A		RCIC-V-12	
MS-V-67B		RCIC-V-13	
MS-V-67C		RCIC-V-19	
MS-V-67D		RCIC-V-22	
MS-V-146		RCIC-V-31	

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TABLE 3.8.4.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
RCIC-V-45	
RCIC-V-46	
RCIC-V-59	
RCIC-V-63	
RCIC-V-64	
RCIC-V-68	
RCIC-V-69	
RCIC-V-76	
RCIC-V-110	
RCIC-V-113	
RFW-V-65A	
RFW-V-65B	
RHR-V-3A	
RHR-V-3B	
RHR-V-4A	
RHR-V-4B	
RHR-V-4C	
RHR-V-6A	
RHR-V-6B	
RHR-V-8	
RHR-V-9	
RHR-V-11A	
RHR-V-11B	
RHR-V-12A	
RHR-V-12B	
RHR-V-16A	
RHR-V-16B	
RHR-V-17A	
RHR-V-17B	
RHR-V-21	
RHR-V-23	
RHR-V-24A	
RHR-V-24B	
RHR-V-26A	
RHR-V-26B	
RHR-V-27A	
RHR-V-27B	
RHR-V-40	
RHR-V-42A	
RHR-V-42B	
RHR-V-42C	
RHR-V-47A	
RHR-V-47B	
RHR-V-48A	
RHR-V-48B	
RHR-V-49	
RHR-V-52A	
RHR-V-52B	
RHR-V-53A	
RHR-V-53B	
RHR-V-64A	
RHR-V-64B	
RHR-V-64C	
RHR-V-68A	
RHR-V-68B	
RHR-V-73A	
RHR-V-74A	
RHR-V-74B	
RHR-V-87A	
RHR-V-87B	
RHR-V-115	
RHR-V-116	
RHR-V-123A	
RHR-V-123B	
RHR-V-124A	
RHR-V-124B	
RHR-V-125A	
RHR-V-125B	
RHR-V-134A	
RHR-V-134B	
RRC-V-16A	
RRC-V-16B	
RRC-V-23A	
RRC-V-23B	
RRC-V-67A	
RRC-V-67B	
RWCU-V-1	
RWCU-V-4	
RWCU-V-31	
RWCU-V-34	

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TABLE 3.8.4.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED	VALVE NUMBER	SYSTEM(S) AFFECTED
RWCU-V-35		SLC-V-1A	
RWCU-V-40		SLC-V-1B	
RWCU-V-42			
RWCU-V-44		SW-V-2A	
RWCU-V-100		SW-V-2B	
RWCU-V-101		SW-V-4A	
RWCU-V-102		SW-V-4B	
RWCU-V-104		SW-V-4C	
RWCU-V-106		SW-V-12A	
		SW-V-12B	
SGT-V-1A		SW-V-24A	
SGT-V-1B		SW-V-24B	
SGT-V-3A1		SW-V-24C	
SGT-V-3A2		SW-V-29	
SGT-V-3B1		SW-V-44	
SGT-V-3B2		SW-V-54	
SGT-V-4A1		SW-V-69A	
SGT-V-4A2		SW-V-69B	
SGT-V-4B1		SW-V-70A	
SGT-V-4B2		SW-V-70B	
SGT-V-5A1		SW-V-75A	
SGT-V-5A2		SW-V-75B	
SGT-V-5B1		SW-V-90	
SGT-V-5B2		SW-V-187A	
		SW-V-188A	
		SW-V-188B	

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ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS power monitoring channels instrumentation shall be determined OPERABLE:

a. At least once per 6 month by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage $\leq$ (132) VAC,

2. Under-voltage $\geq$ (108) VAC, and

3. Under-frequency $\geq$ (57) Hz.

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3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Fuel grapple position.
5. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

---

* See Special Test Exceptions 3.10.1 and 3.10.3.

\# The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

---

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

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3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. At least one with audible indication in the control room,

c. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

d. The "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn# and shutdown margin demonstrations are in progress.

Replace with attached

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS** and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

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

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying that the channel count rate is at least ~~3~~ 0.5 cps:

1. Prior to control rod withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

3. At least once per 24 hours.

replace with attached

d. Verifying, within 8 hours prior to and at least once per 12 hours during, that the RPS circuitry "shorting links" have been removed during:

1. The time any control rod is withdrawn,## or

2. Shutdown margin demonstrations.

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

4.9.2.c) The minimum SRM count rate for operability was reduced to provide maximum flexibility for plant-operational schedules, to allow for operational source removal in subsequent cycles when intrinsic sources provide sufficient counts for operability and to permit the elimination of any potential mechanical failures of unnecessary core components, while maintaining sufficient indication of neutron population levels for core monitoring in the source range. The change is consistent with the recommendations in R.G. 1.68, revision 2.

See also specification 3/4.10.3 justification

attached

3.9.2

d. Adequate assurance of SRM-IRM overlap is established prior to and during the time any control rod is withdrawn# and shutdown margin demonstrations are in progress.

4.9.2

d. Verifying adequate assurance of SRM-IRM overlap has been maintained from the initial overlap demonstration with installed equipment by:

1. confirming no maintenance/replacement of either the SRM or IRM detectors and/or circuitry has modified the respective SRM, IRM scram and rod block setpoints relative to neutron flux levels at the initial overlap demonstration, or

2. With initially installed or modified SRM, IRM detectors/circuitry an overlap demonstration is performed with the RPS circuitry in "non-coincidence mode" ("shorting links" removed) in which the IRMs register neutron flux response above their respective downscale setpoints before the SRMs exceed their respective Rod Block setpoints.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

a. Within 2 hours prior to:

1. The start of CORE ALTERATIONS.
2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.

b. At least once per 12 hours.

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

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3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least (24) hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than (24) hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least (24) hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

DRAFT

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling {platform}~~(floor)~~ personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling {platform} ~~(floor)~~ personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling {platform} ~~(floor)~~ personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of incore instrumentation and control rods with their normal drive system.

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3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

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

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling platform crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds {1200} ± 50 pounds.

b. Demonstrating operation of the overload cutoff on the frame mounted and monorail hoists when the load exceeds 485 ±50 pounds.

c. Demonstrating operation of the normal electrical ~~uptravel mechanical~~ stop on the frame mounted and monorail hoists when uptravel brings the top of (active) fuel assembly to ≥ {8} feet below the {~~normal~~ minimum fuel storage pool} water level.

d. Demonstrating operation of the downtravel mechanical cutoff on the main hoist when grapple hook down travel reaches ≤ 56 feet below track.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than {50} +0 -25 pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds {485 ± 50} pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds {550 ± 50} pounds.

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3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

~~LIMITING~~ CONDITION FOR OPERATION

3.9.7 Loads up to 1500 pounds may be permitted to travel over fuel assemblies in the spent fuel storage pool racks provided that the ~~height of the~~ load above the surface of the pool water is limited per Figure 3.9.7-1. and weight

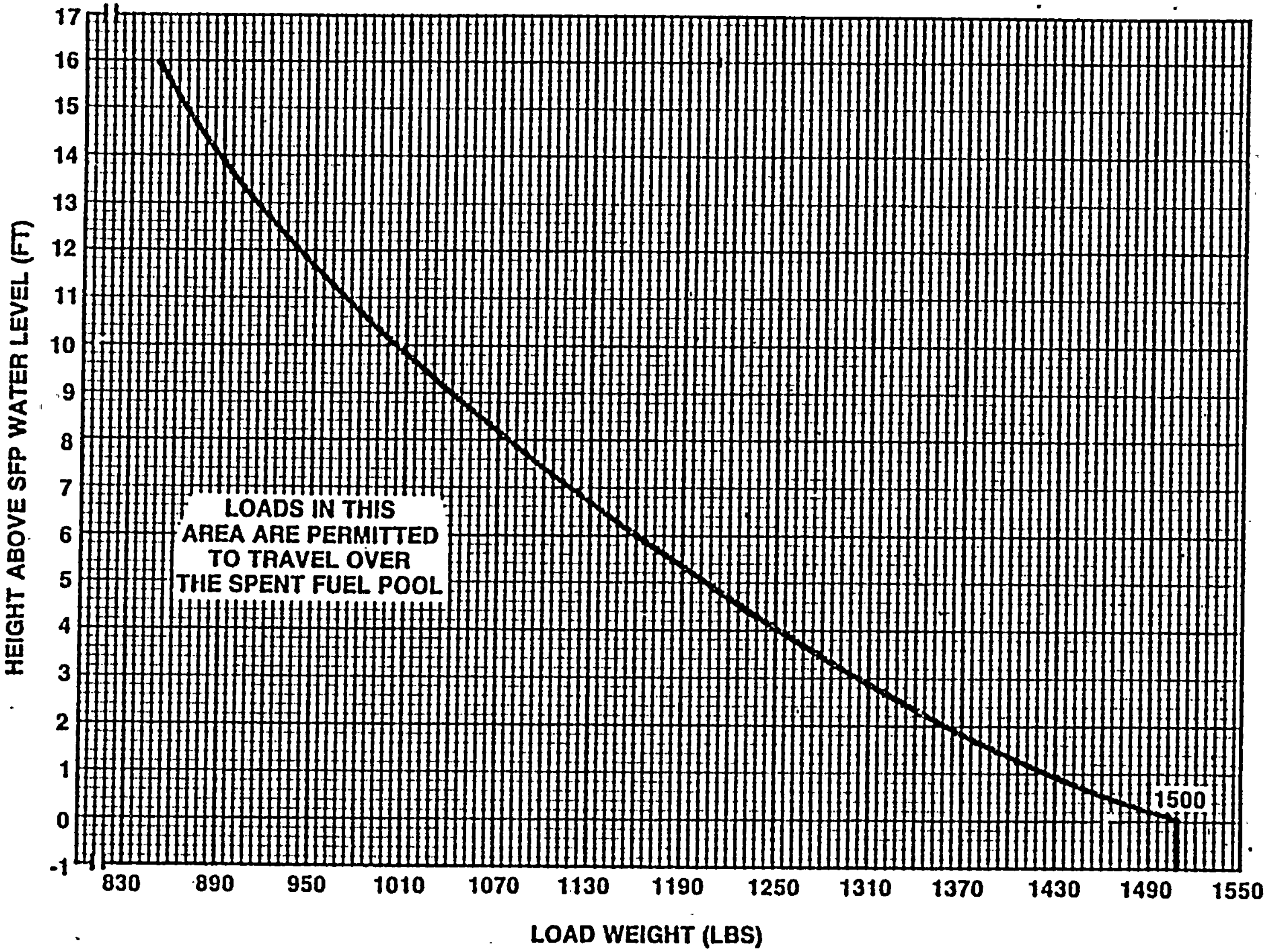
APPLICABILITY: With irradiated fuel assemblies in the spent fuel storage pool racks.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of ~~1500 pounds~~ load limits per Figure 3.9.7-1 over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.



HEIGHT ABOVE SFP WATER LEVEL
VS. MAXIMUM LOAD TO BE CARRIED OVER SFP

FIGURE 3,9.7-1

830381

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3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

(bottom of weir, 605'5" ele.)

3.9.8 At least 22 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

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3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

21'9" (bottom of weir, 605'5" ele.)

3.9.9 At least ~~22~~ feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are ~~in the~~ spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

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3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

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LIMITING CONDITION FOR OPERATION

3.9.10.2. Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

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3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet (bottom of weir, 605'5" ele.) above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

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LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor pressure vessel flange. (bottom of weir, 605'5" ele.)

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternative method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternative method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system, or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

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3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that the rod worth minimizer is OPERABLE per Specifications 3.1.4.1:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RSCS are bypassed, verify;

a. That the RWM is OPERABLE per Specification 3.1.4.1,

b. That movement of control rods from 75 ~~50%~~ ROD DENSITY to the RSCS ~~(preset power level)~~ (low power setpoint) is limited to the approved control rod withdrawal sequence during scram and friction tests,

c. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3, and

d. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

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3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with ~~the RPS circuitry "shorting links" removed~~ adequate assurance of SRM - IRM overlap per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The continuous rod withdrawal control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

Justification

1) The "shorting link" removal requirement was deleted except on initial or post SRM/IRM modification. The combination of IRM range 1 scram and proper SRM/IRM overlap ensures reactor protection system action in time to protect the health & safety of the public from the potential effects of a criticality excursion (which is consistent with the WNP-2 Safety Analysis). The "shorting links" were historically a conservatism applied to initial S/u's when potential SRM saturation could occur prior to reaching trip setpoints from the IRM range instrumentation. Confirmation of overlap/non-saturation once, assures adequate protection until modification requires another check (using the "shorting links").

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched flow may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

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3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.4 may be suspended during the performance of the Startup Test Program until 6 months after initial criticality.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The number of months since initial criticality shall be verified to be less than or equal to 6 months at least once per 31 days during the Startup Test Program.

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3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table 11, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to the concentrations specified in Table 3.11.1.1-1.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11.1.1-1. The results of pre-release analyses shall be used with the calculational methods and parameters in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specifications 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11.1.1-1. The results of the previous post-release analyses shall be used with the calculational methods and parameters in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

Spec. 3/4.11 provided to support related changes.

TABLE 3.11.1.1-1.

MAXIMUM PERMISSIBLE CONCENTRATION OF DISSOLVED OR ENTRAINED NOBLE GASES RELEASED FROM THE SITE TO UNRESTRICTED AREAS IN LIQUID WASTE

NUCLIDE	MPC (CI/ml)*
Kr 84 m	2E-4
Kr 85	5E-4
Kr 87	4E-5
Kr 88	9E-5
Ar 41	7E-5
Xe 133 m	5E-4
Xe 133	6E-4
Xe 135 m	2E-4
Xe 135	2E-4

*Computed from Equation 20 of ICRP Publication 2 (1959), adjusted for infinite cloud submersion in water, and R = 0.01 rem/week, P_W = 1.0 gm/cm^3, and P_W/P_t = 1.0.

TABLE 4.11.1.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (Ci/ml)
Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability, with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and t shall be used in the calculation.

TABLE 4.11.1.1-1

TABLE NOTATION

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

d. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each reactor unit to UNRESTRICTED AREAS (see Figure 5.1.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include the radiological impact on finished drinking water supplies at the nearest downstream drinking water source.

b. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system, as described in the ODCM, shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid waste prior to their discharge when the projected doses due to the liquid effluent, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report which includes the following information:

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

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

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

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each reactor unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 10 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid during the previous 92 days.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to the limits calculated in the ODCM.

APPLICABILITY: At all times.

ACTION:

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

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the temporary tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site (see Figure 5.1.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines and for all radioactive materials in particulate form and radionuclides (other than noble gases) with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to radioactive materials, other than noble gases, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with sampling and analysis program specified in Table 4.11.2-1.

TABLE 4.11.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type *	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Containment Purge	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters f	$1x10^{-4}$
			H-3	$1x10^{-6}$
B. Reactor Building Vents, & Turbine Building Vents	M[b] Grab Sample	M[b]	Principal Gamma Emitters f	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. All Release Types as listed in A and B.	Continuous[e]	W[c, d] Charcoal Sample	I-131	$1x10^{-12}$
	Continuous[e]	W[c, d] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	$1x10^{-11}$
	Continuous[e]	Q Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[e]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$ (XE-133 equivalent)

TABLE 4.11.2-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and t shall be used in the calculation.

TABLE NOTATION

b. If the iodine or particulate monitoring channel(s) is(are) inoperative, analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Particulate and/or charcoal samples shall be analyzed when an alarm is received indicating rate of activity buildup exceeds 3 times normal.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. If the iodine or particulate monitoring channel(s) is(are) inoperative, sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, at the 95% confidence level, together with the above nuclides, shall also be identified and reported.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents from each reactor unit, to areas at and beyond the SITE BOUNDARY see Figure 5.1.1-1 shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from radioiodines and radioactive materials in particulate form, and radionuclides, other than noble gases, with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ, and

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

a. With the calculated dose from the release of radioiodines, radioactive particulates, and radionuclide (other than noble gases) with half-lives greater than 8 days, in gaseous effluent exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2 a Special Report identifying the cause(s) for exceeding the imit and defines the corrective action taken to reduce the releases and assure subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser steam jet air ejector system is in operation.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM inoperable for more than 7 days, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

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

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

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

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM shall be verified to be in operation at least once per 92 days.

RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.5 The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from each reactor unit to areas at and beyond the SITE BOUNDARY (see Figure 5.1.1-1, when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days, or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

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

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

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

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 Doses due to gaseous releases from each reactor unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.2.5 2 The VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 10 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main-condenser offgas treatment system with the hydrogen monitors required OPERABLE by Table 3 3.7.12-1 of Specification 3.3.7.12.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The gross radioactivity (beta and/or gamma) rate of noble gases measured at the main condenser air ejector shall be limited to less than or equal to (3.323 x 10^5 microcuries/sec after 30 minutes decay).

APPLICABILITY: At all times.

ACTION:

With the gross radioactivity (beta and/or gamma) rate of noble gases at the main condenser air ejector exceeding (~~100~~ 3.323 x 10^5 microcuries/sec ~~per MWt~~ after 30 minutes decay), restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases at (near) the outlet of the main condenser air ejector shall be continuously monitored in accordance with Specification 3.3.7.12.

4.11.2.7.2 The gross radioactivity (beta and/or gamma) rate of noble gases from the main condenser air ejector shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the main condenser air ejector.

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

RADIOACTIVE EFFLUENTS

VENTING OR PURGING

LIMITING CONDITION FOR OPERATION

3.11.2.8 VENTING or PURGING of the containment drywell shall be through the primary Containment Vent and Purge System or the Standby Gas Treatment System.

APPLICABILITY: Whenever the drywell is vented or purged.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING and PURGING of the drywell.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11 2 8 The containment drywell shall be determined to be aligned for VENTING or PURGING through the Primary containment Vent and Purge System or the Standby Gas Treatment System within 4 hours prior to start of and at least once per 12 hours during VENTING or PURGING of the drywell.

3/4.11 3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the requirements of 10 CFR Part 20, and/or 10 CFR Part 1, not satisfied. suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

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

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

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes. and

   4. Summary description of action(s) taken to prevent a recurrent.

c. The provisions of Specifications 3.0 3, 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consectuve initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13 to assure SOLIDIFICATION of subsequent batches of waste.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrent of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request. and a variance is not granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3, 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrent. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment or other legitimate reasons. Every reasonable effort shall be made to correct all deficiencies prior to the end of the next sampling period. All deviations will be documented in the annual report.)

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a special report pursuant to Specification 6.9.2. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} \quad \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3.

This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

Spec. 3/4.12 provided to support related changes.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in this table and shown in Figures 3.12-1a and 3.12-1b and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

4.12.1.2 Reports - The results of analyses performed on the radiological environmental monitoring samples shall be summarized in the Annual Radiological Operating Report, pursuant to Specifications 6.9.1.6 and 6.9.1.7.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Sample Type	Sample Location Code[1]*	Sampling and Collection Frequency[1]	Type and Frequency of Analysis[11]
1. AIRBORNE			
a. Particulates and Radioiodine	1, 4-9A, 21, 23, 40, and 48	Continuous Sampling Weekly Collection	Particulate: Gross beta[2] Gamma isotopic[3] on quarterly composite (by location) Radioiodine: Gamma for I-131 weekly
2. DIRECT RADIATION[4]	1-9A, 10-25, 40-46, 1S-16S	Quarterly	Gamma Dose
3. WATERBORNE			
a. Surface	26 and 27	Composite aliquots monthly[5]	Gamma isotopic[3]
b. Drinking Water	28 and 29	Composite aliquots monthly[5]	Gamma isotopic[3]
c. Ground Water	--	7	7
d. Sediment from Shoreline	33 and 34	Semi-annually	Gamma isotopic[3]

TABLE 3.12-1
(Continued)

Sample Type	Sample Location Code[1]*	Sampling and Collection Frequency[1]	Type and Frequency of Analysis[11]
4. INGESTION			
a. Milk[8]	9C, 35, 36, and 40	Semi-monthly during grazing season Monthly at other times	Gamma isotopic[3] Iodine - 131
b. Fish[9]	30, 38 and 39	Semi-annually, 4 in vicinity of discharge 1 from Snake River	Gamma isotopic[3] on edible portions
c. Garden Produce[10]	37A and 37B 9B	Monthly during growing season in the River-view area of Pasco; and a control at Grandview	Gamma isotopic[3] on edible portions

*Sample locations are shown on Figures 3.12-1a and 312-1b.

1 Deviations are permitted if samples are unobtainable due to hazardous conditions, seasonal availaility, malfunction of automatic sampling equipment, or other legitimate reasons. All deviations will be documented in the annual report.

2 Particulate sample filters will be analyzed for gross beta after at least 24 hours decay. If gross beta activity is greater than 10 times the mean of the control sample, gamma isotopic analysis should be performed on the individual sample.

3 Gamma isotopic means identification and quantification of gamma emitting radionuclides that may be attributable to the effluents of the facility.

4 TLDs used in the REMP meet the requirements of ANSI N545-1975 except for the energy dependence (Section 4.3.4) specified for the region between 30-50 kev. TLD locations 1S-16S are sampled to demonstrate compliance with the restricted boundary dose limits.

5 Composite samples will be collected with equipment which is capable of collecting an aliquot at time intervals which are short relative to the compositing period.

TABLE 3.12-1
(Continued)

6 Tritium analysis will be performed on a quarterly composited sample.

7 Ground water samples are not collected since there are no wells which are presently being used for either drinking or irrigation where the hydraulic gradient or recharge properties are suitable for contamination by WNP-2 effluents.

8 Milk samples will be obtained from farms or individual milk animals which are located in sectors with the higher calculated annual average ground-level D/Q's. If Cesium-134 or Cesium-137 is measured in an individual milk sample in excess of 30 pCi/l, then Strontium 90 analysis should be performed.

9 The species of interest to be collected are primarily salmonids.

10 Garden produce will be obtained from farms or gardens which use Columbia River water, if possible, for irrigation and different varieties will be obtained as they are in season. One sample of root food, leafy vegetables, and fruit should be collected each period.

11 Frequency of analysis will be as collected or as stated in these footnotes for special case.

TABLE 3.12-2

Station	Sector	Radial Miles(a)	Direct Radiation	Air Particulate & Radioiodines	Surface Water	Drinking Water	Groundwater	Shoreline Sediment	Milk	Fish	Garden Products
1S	N	0.3	X								
2S	NNE	0.4	X								
3S	NE	0.5	X								
4S	ENE	0.4	X								
5S	E	0.4	X								
6S	ESE	0.4	X								
7S	SE	0.5	X								
8S	SSE	0.7	X								
9S	S	0.7	X								
10S	SSW	0.8	X								
11S	SW	0.7	X								
12S	WSW	0.5	X								
13S	W	0.5	X								
14S	WNW	0.5	X								
15S	NW	0.5	X								
16S	NNW	0.4	X								
1	S	1.3	X	X							
2	NNE	1.8	X								
3	SE	2.0	X								
4	SSE	9.3	X	X							
5	ESE	7.7	X	X							
6	S	7.7	X	X							
7	WNW	2.7	X	X							
8	ESE	4.7	X	X							
9A*	WSW	30.0	X	X							
9B*	WSW	35.0									X
9C*	WSW	33.0							X		
10	E	3.1	X								
11	ENE	2.0	X								
12	NNW	6.1	X								
13	SW	1.4	X								
14	WSW	1.4	X								
15	W	1.4	X								
16	WNW	1.4	X								
17	NNW	1.2	X								
18	N	1.1	X								
19	NE	1.8	X								
20	ENE	1.9	X								

*Control Location

TABLE 3.12-2

Station	Sector	Radial Miles(a)									
21	ENE	1.0	X	X							
22	E	2.1	X								
23	ESE	3.0	X	X							
24	SE	1.9	X								
25	SSE	1.6	X								
26*	E	3.2			X	X[d]					
27	E	3.2			X						
28	SSE	7.4				X					
29	SSE	11.0				X					
30	E	3.2								X	
31(b)											
32(b)											
33*	ENE	3.3							X		
34	ESE	3.3							X		
35	ENE	10.5								X	
36	ESE	7.2								X	
37A	SSE	17.0									X
37B	SSE	17.0									X
38*	E	93.0								X	
39	NE	4.3								X	
40	SE	6.4	X						X		
41	SE	5.8	X								
42	ESE	5.6	X								
43	E	5.7	X								
44	ENE	5.7	X								
45	ENE	4.2	X								
46	NE	4.7	X								
48	NE	4.3		X							

* Control Location
(a)Taken from center of WNP-2 containment.
(b)Not currently used.
(c)Not currently sampled.
(d)Intake water (station 26) is also the control location for drinking water.

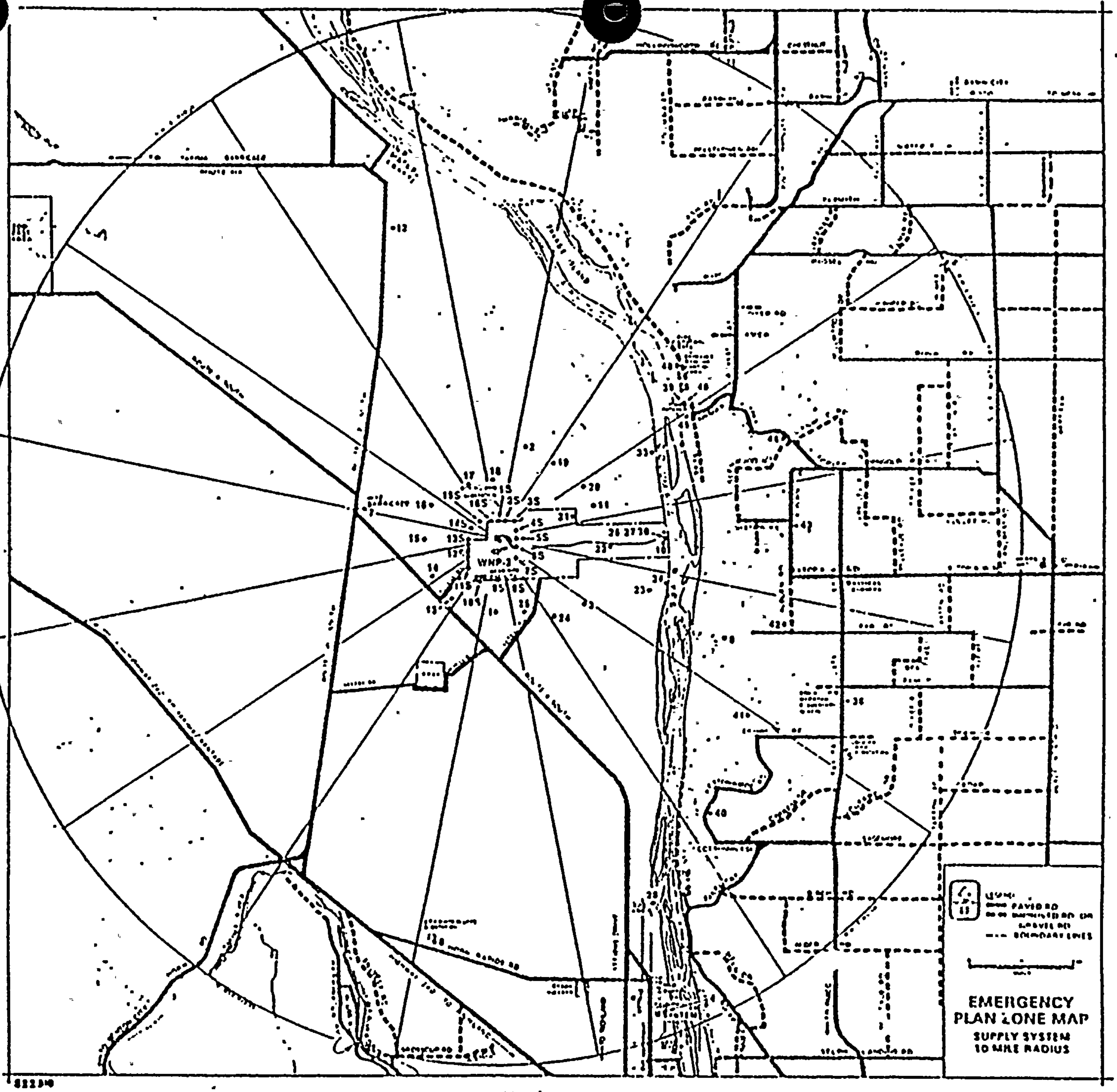

FIGURE 3.12-1(a)
RADIOLOGICAL ENVIRONMENTAL MONITORING SAMPLE LOCATIONS INSIDE OF 10 MILE RADIUS

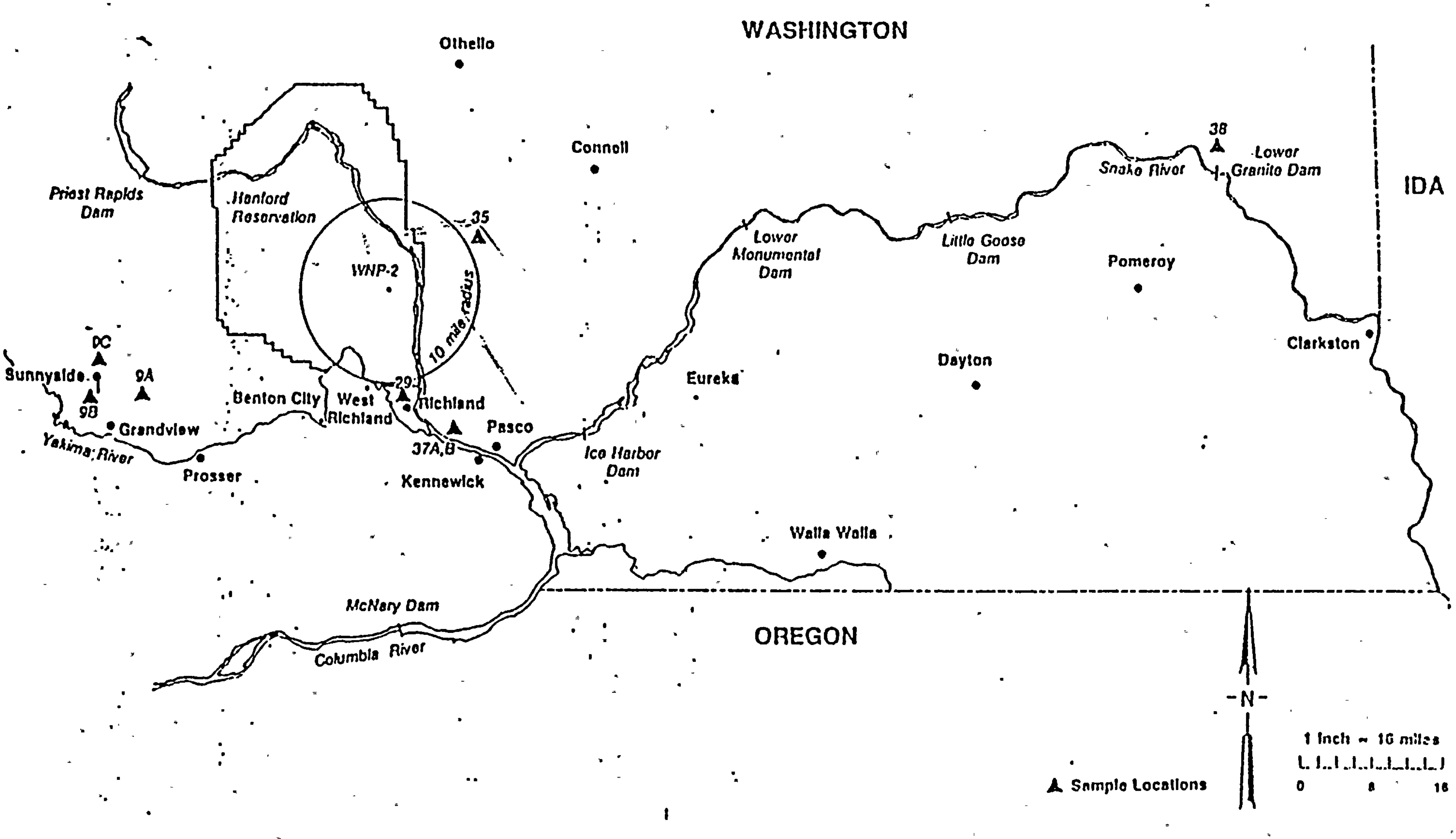

FIGURE 3.12 - 1(b) Radiological Environmental Monitoring Sample Locations Outside of 10-Mile Radius

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)
H-3	2×10^{4} (a)				
Mn-54	1×10^{3}		3×10^{4}		
Fe-59	4×10^{2}		1×10^{4}		
Co-58	1×10^{3}		3×10^{4}		
Co-60	3×10^{2}		1×10^{4}		
Zn-65	3×10^{2}		2×10^{4}		
Zr-Nb-95	4×10^{2} (b)				
I-131	2	0.9		3	1×10^{2}
Cs-134	30	10	1×10^{3}	60	1×10^{3}
Cs-137	50	20	2×10^{3}	70	2×10^{3}
Ba-La-140	2×10^{2} (b)			3×20^{2} (b)	

(a) For drinking water. This is 40 CFR Part 141 value.
(b) Total for parent and daughter.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMIT OF DETECTION (LLD)[a]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, dry)	Sediment (pCi/Kg, dry)
gross beta	4[b]	1×10^{-2}				
^{3}H	2000[b]					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	1[b]	7×10^{-2}		1	60[c]	
^{134}Cs	15	5×10^{-2}	130	15	60	150
^{137}Cs	18	6×10^{-2}	150	18	80	180
^{140}Ba	60			60		
^{140}La	15			15		

Note: This list does not mean that only these nuclides are to be detected and reported. Other peaks which are routinely measurable and identifiable shall also be identified and reported.

[a]The LLD is the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

TABLE 4.12-1 (Continued)

For a particular measurement system (which may include radiochemical separation):

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda t)}$$

where,

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume).

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficienty (as counts per disintegration).

V is the sample size (in units of mass or volume).

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

t is the elapsed time between sample collection and counting

The value of s_b used in the calculation of the LLD for a particular measurement system should be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background should include the typical contributions of other radionculides normally present in the samples (e.g., potassium-40 in milk samples).

Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLD's unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

[b]LLD for drinking water.

[c]LLD for leafy vegetable.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) significantly greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1 prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location, if available, shall be added to the radiological environmental monitoring program as soon as practicable. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2.1 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1, by door-to-door survey, aerial survey, or by consulting local agriculture authorities using that information which will provide the best results.

4.12.2.2 Reports - The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

3/4.12.3 INTER-LABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 The laboratories of the Supply System or its contractors which perform analyses shall participate in the Environmental Protection Agency's (EPA's) Environmental Radioactivity Laboratory Intercomparisons Studies (Crosscheck) Program or equivalent program which has been approved by the Commission. This participation shall include all of the determinations (sample medium-radionuclide combination) that are offered by EPA and that are also included in the monitoring program. The results of analysis of these crosscheck samples shall be included in the annual report.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Inter-laboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operation Report. As an alternative, the Supply System laboratories or those of its contractors participating in the EPA crosscheck program may provide their EPA program code so that the NRC can review the EPA's participant data directly in lieu of submission in the annual report.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

DRAFT

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 requires two control room emergency filtration subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours. As a further example, Specification 3.6.6.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. & at least cold shutdown ~~in~~ within the following 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.03) change made to comply with previous example given.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL CONDITIONS or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance; instead, it permits the more frequent performance of surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that surveillance activities associated with a Limiting Conditions for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

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4.0.5 This specification ensures that 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 will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements ~~has been~~ will be provided in writing ~~by~~ to the Commission and ~~is not~~ will become a part of ~~these Technical Specifications.~~ the inservice inspection program.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

Justification:

10 CFR 50.55a.g.5.iv provides for requesting relief where an examination or test is impractical. The request will be submitted to the NRC for evaluation and any relief provided will become a part of the inservice inspection program. The change more accurately reflects 10 CFR 50 which is the justification for the change.

DRAFT

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + {0.38}% delta k/k or R + {0.28}% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the HUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by {an insequence} control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

the fuel cladding safety limit

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than ~~[illegible]~~ during the limiting power transient analyzed in Section 15.2 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than ~~[illegible]~~. The occurrence of scram times longer then those specified should be viewed as an indication of a systematic ~~systemic~~ problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

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CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than {3} inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than {20}% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section (15.4.9) of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

DRAFT

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 ~~(3420)~~ gallons of solution containing a minimum of 5500 pounds of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of [150] ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required minimum pumping rate is [41.2] gpm. The minimum storage volume of the solution is esablished to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement on the sodium pentaborate solution is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's", G. E. Topical Report NEDO-10527, March 1972

2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972

3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores", Supplement 2 to NEDO-10527, January 1973

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times $\leq 1.02 \geq$ is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3.

Insert attachment

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

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BASES 3/4 2.1, PAGE B3/4 2-1

2nd Paragraph: Insert following:

"...is shown in Figure 3.2.1-1, 3.2.1-2 and 3.2.1-3 for

...~~is shown in Figure 3.2.1-1 for~~ two recirculation loop operation. These values are to be multiplied by a factor of (later) for single loop operation. The calculational procedures and significant input parameters are documented in FSAR Section 6.3.3. The reduction factor derived for single-loop operation is justified in an amendment to the FSAR.

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AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.49 for 8 x 8 fuel. The flow biased simulated thermal power-upscale scram setting and flow biased neutron flux-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.06 or that ≥ 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and peak flux indicates a TOTAL PEAKING FACTOR greater than 2.49. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

Replace with (over)

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and flow biased neutron flux-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that $\geq$ 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY, CMFLPD, indicates a higher peaked power distribution to ensure that a LHGR transient would not be increased in the degraded condition.

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Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... 3462 Mwt* which corresponds to {105}% of rated steam flow

Vessel Steam Output .... 15.01×10^6 lbm/hr which corresponds to {105}% of rated steam flow

Vessel Steam Dome Pressure............... 1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 3.1, 1.0 ft^2

b. Small Breaks 0.1 ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	~~1.18~~ ~~1.20~~

leave as is.

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsections 6.2, 6.3 and 15.6 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR ~~of 1.06~~, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR ~~of 1.06~~, the required minimum operating limit MCPR of Specification 3.2.3 is obtained. ~~(and presented in Figure 3.2.3-1)~~.

15.0-2

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table ~~5.2-2~~ that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-2 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated. The K_f factors were derived using THERMAL POWER and core flow corresponding to 105% of rated steam flow.

The K_f factors were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

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MINIMUM CRITICAL POWER RATIO (Continued)

~~The K_f factors shown in Figure 3.2.3-1 are conservative for the General Electric plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.~~

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November, 1975.

2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.

3. Qualification of the One Dimensional Core Transient Model For Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program For The Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

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3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

~~3/4.3.2~~ ~~ISO~~LATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing ~~the OPERABILITY trip~~ setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a ~~≤~~3~~½~~ second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of ~~10~~ 12 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the ~~10~~ 13 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the ~~10~~ 13 second delay. It follows that checking the valve speeds and the ~~10~~ 12 second time for emergency power establishment will establish the response time for the isolation functions. ~~However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.~~ Removed per R. Bottimore as a result of LRG-I effort.

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.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

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.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979~~, and Appendix (___) of the FSAR).~~

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine throttle valves and fast closure of the turbine governor valves.

A fast closure sensor from each of two turbine governor valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine governor valves provides input to the second EOC-RPT system. Similarly, a {position switch} for each of two turbine throttle valves provides input to one EOC-RPT system; a {position switch} from each of the other two throttle valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine governor valves and a 2-out-of-2 logic for the turbine throttle valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the {time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., {190}ms, less the time allotted for sensor response, i.e., {10}ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., {83}ms, and plant pre-operational test results}.

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.

delete → replace with: (over)

Replacement:

Operation with one or two EOC-RPT trip systems inoperable is acceptable on the basis that the required operating limit CPR analyses in FSAR chapter 15 accounts for no EOC-RPT and that the MCPR limit is established at a more limiting value, thereby accounting for the lack of EOC-RPT.

DRAFT

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

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.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls ~~and~~ Section 3/4.2 Power Distribution Limits, and 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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~~ an allowance for instrument drift specifically allocated for each trip in the safety analyses.

per GE

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of (NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980).

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. {This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.}

DRAFT

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.7.8 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection system ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", February 1975 ~~(Revision 1, January, 1977)~~.

DRA

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warrented for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficien capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. (The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.)

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.~~8~~9 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure.

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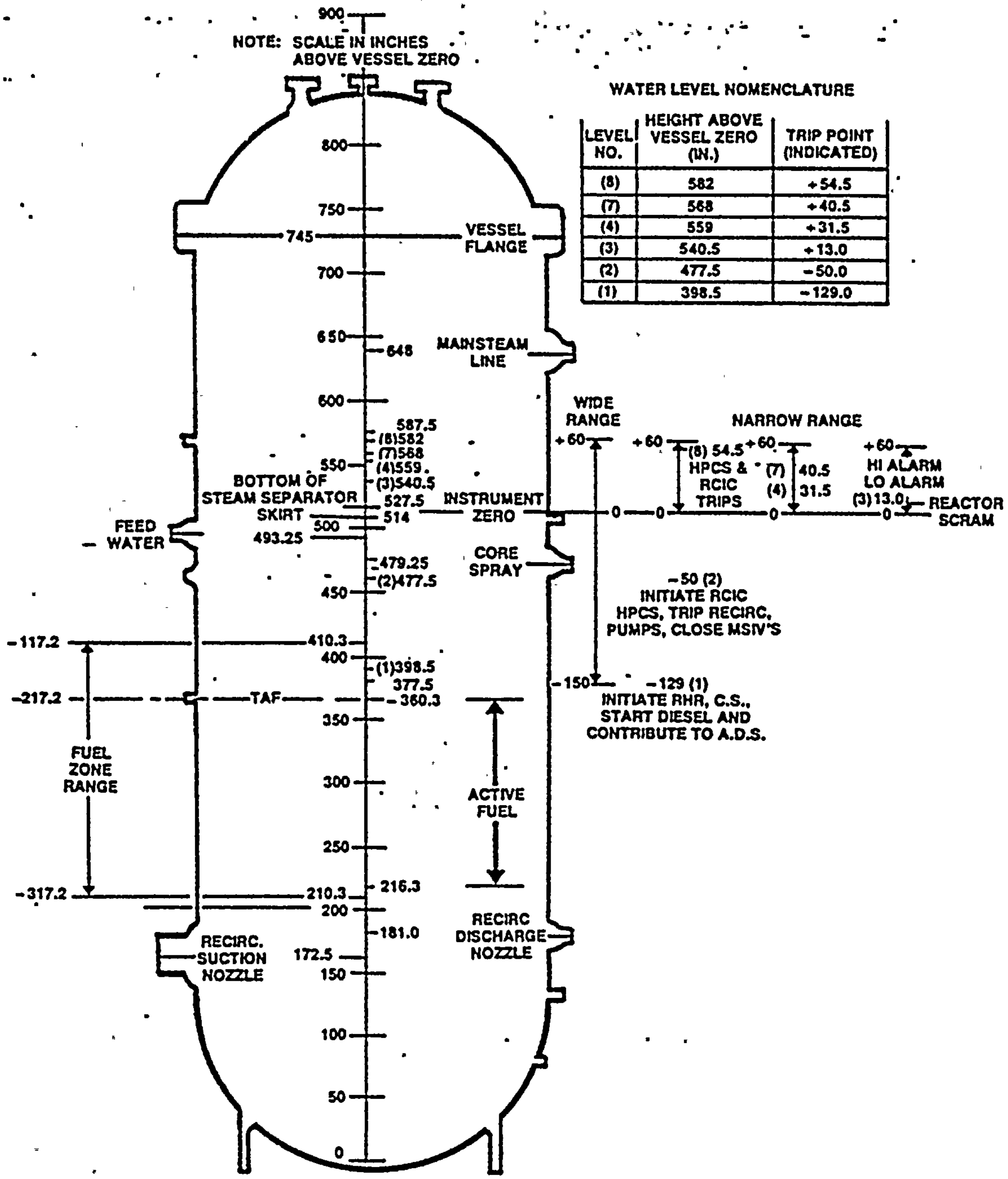
3/3.3.7.11 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/3.3.7.12 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.



WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (IN.)	TRIP POINT (INDICATED)
(8)	582	+54.5
(7)	568	+40.5
(4)	559	+31.5
(3)	540.5	+13.0
(2)	477.5	-50.0
(1)	398.5	-129.0

BASES FIGURE B3/4.3-1. REACTOR VESSEL WATER LEVEL

822064.15A

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Insert attached

~~Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.~~

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria. [The limits will ensure an adequate core flow coastdown from either recirculation ~~loop~~ loop following a LOCA.]

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety-relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of ~~(13)~~ 12 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

delete, N/A

~~The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of (7) valves and lowering the opening setpoint of (2) valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.~~

Bases Section 3/4.4.1, Page B3/4 4-1

Replace first paragraph of Bases Section 3/4.4.1, Page B3/4 4-1 with:

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR fuel cladding safety and operating limits are increased as noted by Specification 2.1.2 and 3.2.3, APRM scram, rod block and RBM Setpoints are adjusted as noted in Specification 3.2.2, Table 2.2.1-1 and Table 3.3.6.2, MAPLHGR limits are decreased by the factor given in Specification 3.2.1, and ~~operating~~ recirculation ~~pump speed or recirculation~~ flow control valve position is reduced as noted in Specification 3.4.1.1.

N/A for WNP-2

-Change is result of GE anlayses (plant specific), allowing single loop operation.

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3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

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

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping (i.e., those that are subject to high stress or that ~~contain~~ contain relatively stagnant, intermittant, or low flow fluids) ~~requires additional surveillance, and leakage limits.~~ is included in ISI Program Plans (cf. Technical Specification 4.4.8).

ADDITIONAL SURVEILLANCE IS

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. ~~Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.~~

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam line rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence~~, as well as adjustments for possible errors in the pressure and temperature sensing instruments~~.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 1.

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PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

access

~~Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.~~

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

Access for inservice inspection of reactor coolant system components is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition with Addenda through Summer, 197

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BASES TABLE B 3/4.4.6-1

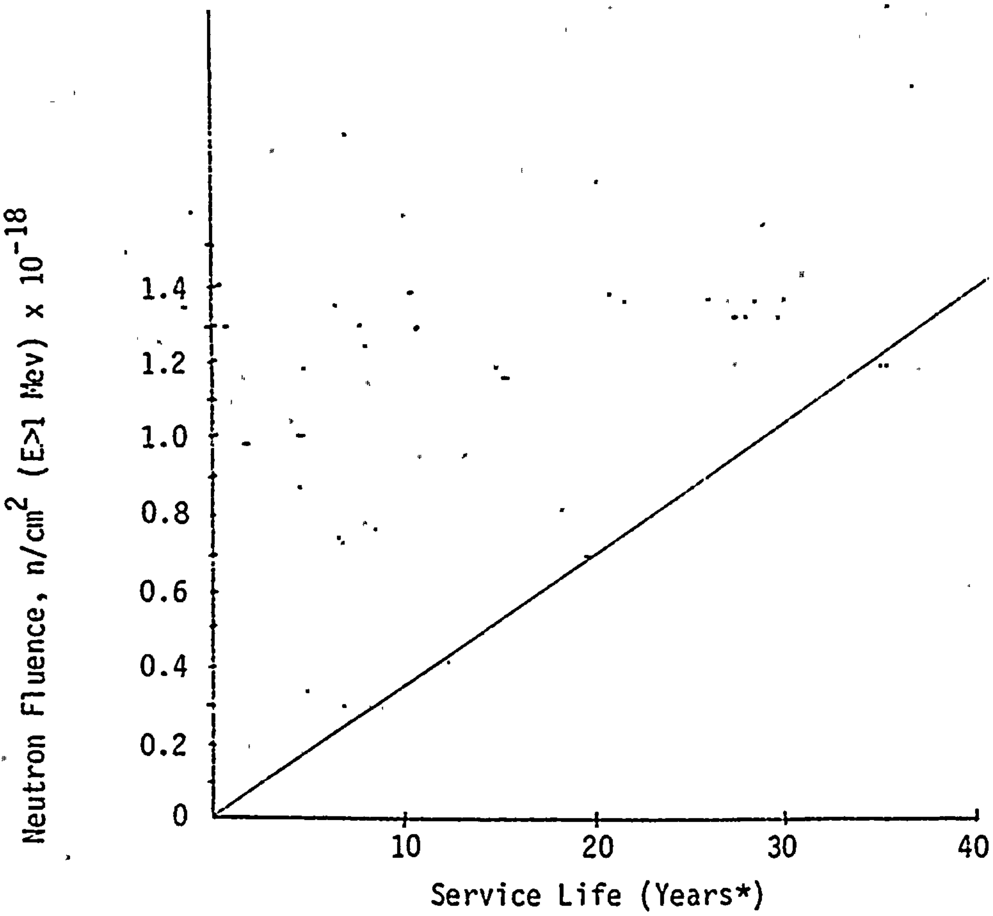
REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %*	P %*	DROPWEIGHT RT_{NDT} °F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS**	ΔRT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS**
BELTLINE										
RING 1 PLATE		SA-533 GRB CL1	.15	.014	-10	+28		45	>100	
RING 2 PLATE		SA-533 GRB CL1	.15	.018	-30	-8		52	>100	
GIRTHWELD		E8018 NM	.03	.020	NA	-50		33		
		RACO1 NMM	.08	.016	NA	-44		27		
NON BELTLINE										
RING 3 PLATE		SA-533 GRB CL1								
RING 4 PLATE		SA-533 GRB CL1								
VESSEL FLANGE		SA-508 CL 2								
TOP HEAD FLANGE		SA-508 CL 2								
TOP HEAD DOLLAR PLATE		SA-533 GRB CL1								
TOP HEAD SIDE PLATES		SA-533 GRB CL1								
BOTTOM HEAD DOLLAR PLATES		SA-533 GRB CL 1								
BOTTOM HEAD RADIAL PLATES		SA-533 GRB CL1								
NOZZLES		SA-508 CL2								
FLANGE BOLT STUDS		SA-540 B23								

NA - NOT AVAILABLE

* HIGHEST VALUE OF ALL ITEMS

** TRANSVERSE SPECIMENS NOT REQUIRED, SEE WNP-2 FSAR 5.3.1.5



Fast Neutron Fluence (E>1 Mev) at ¼T As a Function of Service Life*

Bases Figure B 3/4.4.6-1

* At 90% of RATED THERMAL POWER and 90% availability

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3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping.~~and to start cooling at the earliest moment.~~

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping~~, and to start cooling at the earliest moment.~~

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of ~~1130~~ 1160 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

ECCS-OPERATING and SHUTDOWN (Continued)

1160/1130/200 516/1550/6350

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to (~~1550/1650/6250~~) gpm at (~~reactor~~) {differential} pressures of (~~1130/1110/200~~) psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, {but no credit is taken in the safety analyses for the condensate storage tank water}.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, {a system for which no credit is taken in the safety analysis}, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds {100} psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety-relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

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SUPPRESSION CHAMBER (Continued)

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

~~In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume and vortex prevention plus a (2'-4') safety margin for conservatism.~~

Delete; lower pool level analysis performed by A/E but level not used in tech spec's by WNP-2

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 34.7 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks. ~~after each opening.~~

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

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3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 34.7 ~~psig in the event of a~~ LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure ~~of 34.7 psig~~ does not exceed the
45 design pressure of ~~[illegible]~~ psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure
2 differential of ~~[illegible]~~ psid. ~~The limit of 1.75 psig for initial positive containment pressure will limit the total pressure to 34.7 psig which is less than the design pressure and is consistent with the safety analysis.~~

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

See Next page for Revised section

The 24 inch and 30 inch drywell and suppression chamber purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the purge system. To provide assurance that the 24 inch and 30 inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4, which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.

The use of the drywell and suppression chamber purge lines is restricted to the 2 inch purge supply and exhaust isolation valves since, unlike the 24 inch and 30 inch valves, the 2 inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The design of the 2 inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

(Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.)

3/4 6.1.8 DRYWELL AND SUPP. CHAMBER PURGE SYSTEM

While the containment purge system provides plant operational flexibility, its design and use must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant operation must not rely on its use on a routine basis.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. ~~(The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.)~~

Delete: This is a statement ab[out] the pump and valve test program, not a basis.

Justification for above changes:

The subject valves have been qualified per the requirements of the WNP-2 SER section 6.2.4.3 and are capable of closing during a steamline break accident, or LOCA. See section 3.6.1.8 for other details.

3/4.6.1.6 • Peak pressure is defined in Tech. Spec 3/4.6.1.2 and in the FSAR.
• Statements are essentially redundant. Delete 2nd statement.

DRAFT

3/4.6.1.9 PRIMARY CONTAINMENT PENETRATION PRESSURIZATION ~~SYSTEM~~

The OPERABILITY of the primary containment penetration pressurization system is required to meet the restrictions on overall containment leak rate assumed in the accident analyses. The Surveillance Requirements for determining OPERABILITY are consistent with Appendix "J" of 10 CFR 50.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that ~~the primary containment~~ pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately ~~35~~ 34.7 psig which is below the design pressure of 45 psig. Maximum water volume of 827 ~~128,360~~ ~~ft³~~ results in a downcomer submergence of 12'~~3"~~ and the minimum volume of 197 ~~127,080~~ ~~ft³~~ results in a submergence approximately four inches less. The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and ~~Bogeda~~ Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately ~~142~~ 135°F immediately following blowdown which is below the 200°F used for complete condensation via quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

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DEPRESSURIZATION SYSTEMS (Continued)

Experimental data indicates that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below ~~205°F~~ 200°F during any period of relief valve operation with sonic conditions at the discharge exit for quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. ~~(Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.)~~ Per GE

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valv inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment ~~(and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50)~~. Containment isolation within the time limits specified ensures for those isolation valves designed to close automatically that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the Reactor Building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are ~~(four pairs)~~ nine of valves to provide redundancy & capacity. ~~so that operation may continue for up to 72 hours with no more than one pair of vacuum breakers inoperable in the closed position.~~

BASIS: FSAR addresses compliance with GDC 54-57 of 10CFR50, App A.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. Either drywell and suppression chamber hydrogen recombiner system is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. ~~(The drywell (and suppression chamber) hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.)~~ The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", September 1976.

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3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

3/4.7.4 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed, nominal time less 25%, may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those snubbers which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Failures of these sample snubbers may require functional testing of additional units.

This section re-written to reflect 3/4.7.4 rewrite.

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4 7.6 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinkler systems, CO_2 systems, Halon systems and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the {weight and pressure} ~~(level)~~ of the tanks. ~~(Level measurements are made by either a U.L. or F.M. approved method.)~~ WNP-2 option.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

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3/4.7.7 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. ~~The temperature limits include allowance for an instrument error of 1°F.~~ N/A for WNP-2

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the ~~feedwater controller failure~~ analysis of FSAR Chapter 15.1. The bypass system along with the Safety Relief Valves prevent over pressurization of the reactor ~~pressurization~~ pressure vessel.

Justification: Change made to reflect FSAR chapter 15.1 analysis.

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3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least Division 1 or 2 of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. or D.C. source. Division 3 supplies the high pressure core spray (HPCS) system only.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When diesel generator (1) or (2) is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator (1) or (2) as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period diesel generator (1) or (2) is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977 and Regulatory Guide 1.137" Fuel-Oil Systems for Standby Diesel Generators", Revision 1, October 1979.

FEB [illegible]

A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants", February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8.2.1-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

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BASES

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are than tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The ~~(bypassing) (or)~~ OPERABILITY of the motor operated valve thermal overload protection ~~(continuously) (or) (during accident conditions) (by integral bypass devices)~~ ensures that the thermal overload protection ~~(during accident conditions)~~ will not prevent safety related valves from performing their function. The surveillance requirements for demonstrating the ~~(bypassing) (or)~~ OPERABILITY of the thermal overload protection ~~(continuously) (and) (or) (during accident conditions)~~ are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977).

Plant specific changes

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3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

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3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the ~~refueling platform will~~ be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals ~~and pressure vessel~~ are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL - SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 22 feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 22 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

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3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

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

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limits for dissolved or entrained noble gases were determined by converting their MPC's in air to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achieveable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operation, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

BASES

3/4.11.1.3 LIQUID WASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonbly achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the dose associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)(1)). For individuals who may at times be within the SITE BOUNDARY, the occupancy of the individual will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding throid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of gaseous effluents from all reactors at the site.

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man form Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

The specificiation is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of

3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)

Appendix I that conformance with the guides of Appendix I be shown by calculational procedures bsed on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.11, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive materials in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 AND 3.4.11.2.5 GASEOUS RADWASTE TREATMENT SYSTEM AND VENTILATION EXHAUST TREATMENT SYSTEM

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

The specification is provided to ensure that concentration of potentially explosive gas mixtures contained in the off gas treatment system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Deisgn Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MAIN CONDENSER

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.2.8 VENTING OR PURGING

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas.

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing the packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 CFR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR 190 limts. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions frm other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations or radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defines as an "a priori" (before the fact) limit representing the capability of a measurement system and not as "a posteriori" (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operation Report.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity 26 kg/year of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

3/4.12.3 INTER-LABORATORY COMPARISON PROGRAM

The requirement for participation in an Inter-laboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

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SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

/Site

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

(OVER) 5.1.3

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel pressure vessel consisting of a drywell and suppression chamber. The drywell is a steel-lined prestressed concrete vessel in the shape of a truncated cone on top of a water filled suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 200,540 cubic feet. The suppression chamber has an air region of 144,184 cubic feet and a water region of 137,262 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 45 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool [275]°F.

c. Maximum external pressure 2 psig.

d. Maximum floor differential pressure: 25 psid, downward.
6.4 psid, upward.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, the Reactor Building recirculation fan room, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 3,500,000 cubic feet.

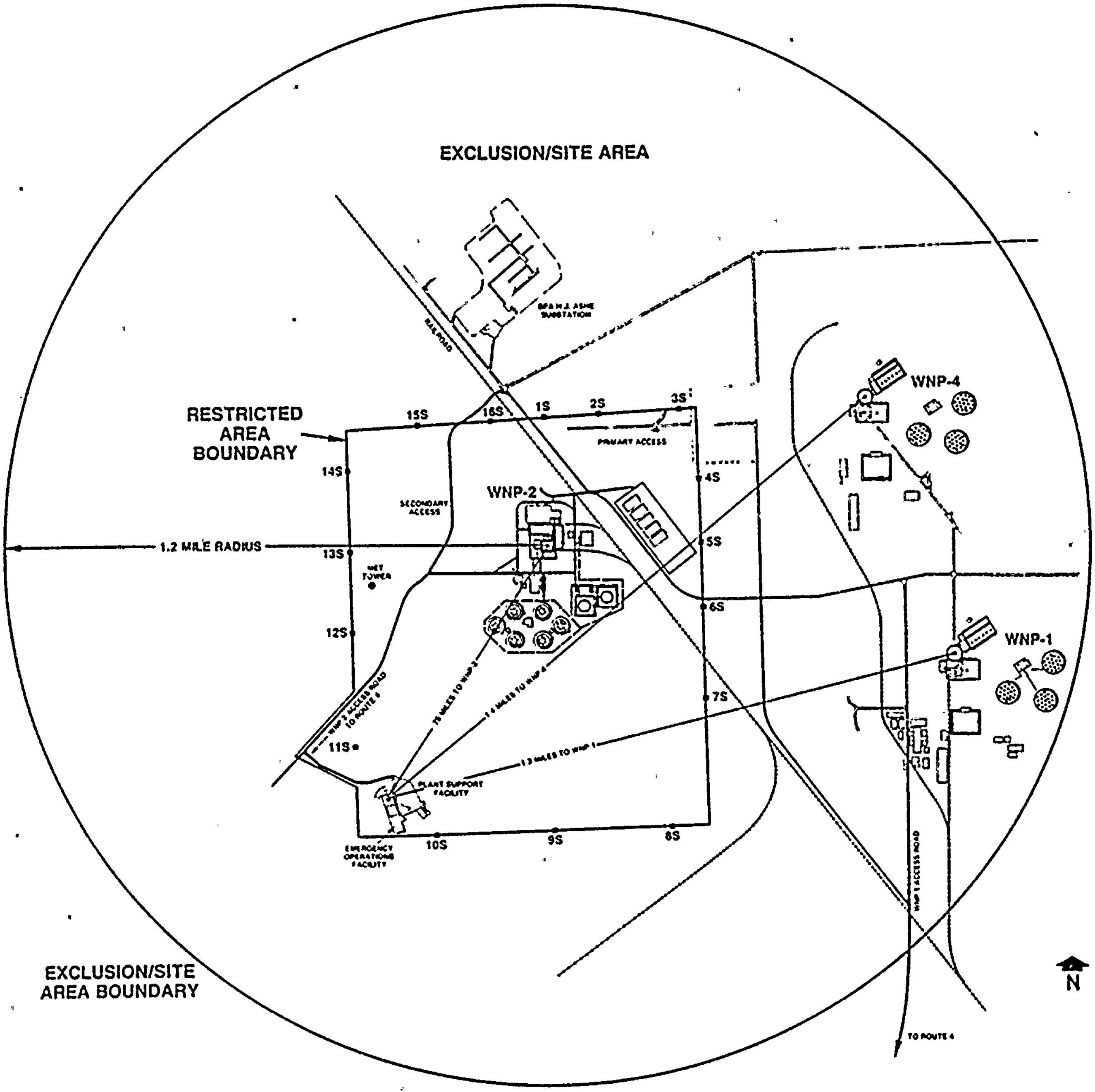
Add :

Restricted Area :

5.1.3 The Restricted Area ~~will~~ shall be ~~shown~~ as shown in figure 5.1.1-1.

Justification :

Changes reflect terminology used in environmental tech. spec. sections and will lessen confusion.



EXCLUSION/SITE AND
RESTRICTED AREA BOUNDARIES

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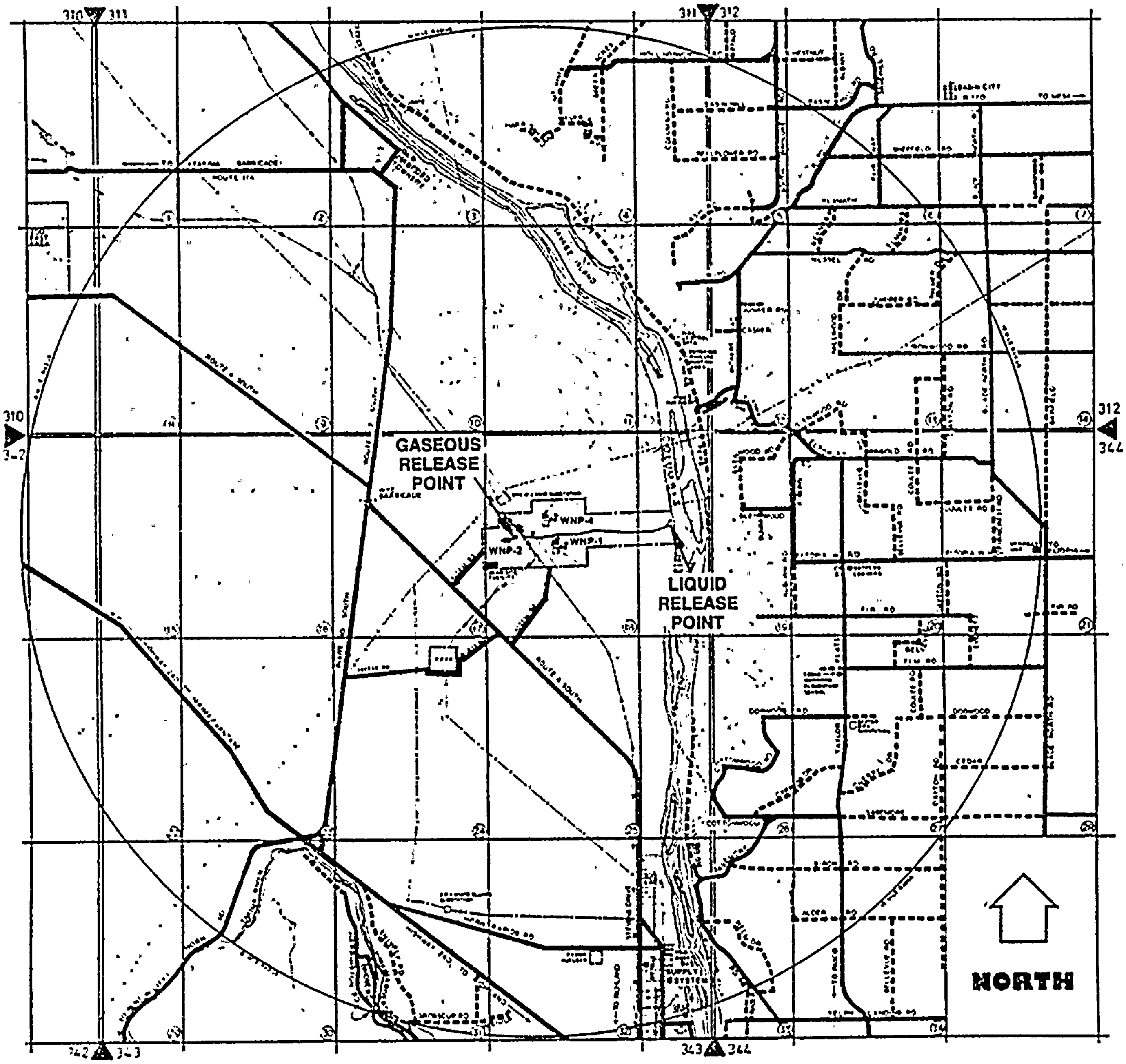
FIGURE 5.1.1-1



LOW POPULATION ZONE

WASHINGTON NUCLEAR — UNIT 2

FIGURE 5.1.2-1

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading. ~~and shall have a maximum average enrichment of ( ) weight percent U-235.~~

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately ~~21,670~~ 22,539 cubic feet at a nominal steam dome saturation temperature of ~~547~~ 545°F.

5.3.1 The supply system has not purchased reload fuel and is therefore unable to provide this info. now, nor is it predicted to be available prior to fuel load of WNP-2.

5.4.2. Change made per GE review

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6. FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section ~~(4.3)~~ 9.1.2 of the FSAR.

b. A nominal 6.5 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooding with water is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 605'7".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2658 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	~~120~~ 117 heatup and 111 cooldown cycles	~~70~~ 100°F to 560°F to ~~70~~ 100°F
	80 step change cycles	Loss of ~~(all)~~ feedwater heaters
	{180} reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 → ~~{40}~~ hydrostatic pressure ~~(and leak tests)~~	Pressurized to ~~≥ {930} psig~~ ~~>0~~ ≤ {1250} psig
	123 Bolt up cycles	Operations Cycle (at 70°F)

Changed per GE dwg. 762E120 Rev. 2

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SECTION 6.0

ADMINISTRATIVE CONTROLS

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

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6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Director of Power Generation shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figures 6.2.2-1a and 6.2.2-1b, and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room.

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least five members shall be maintained on site at all times*. The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, nor the three other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Health Physics Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

~~[The amount of overtime worked by unit staff members performing safety related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).]~~

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

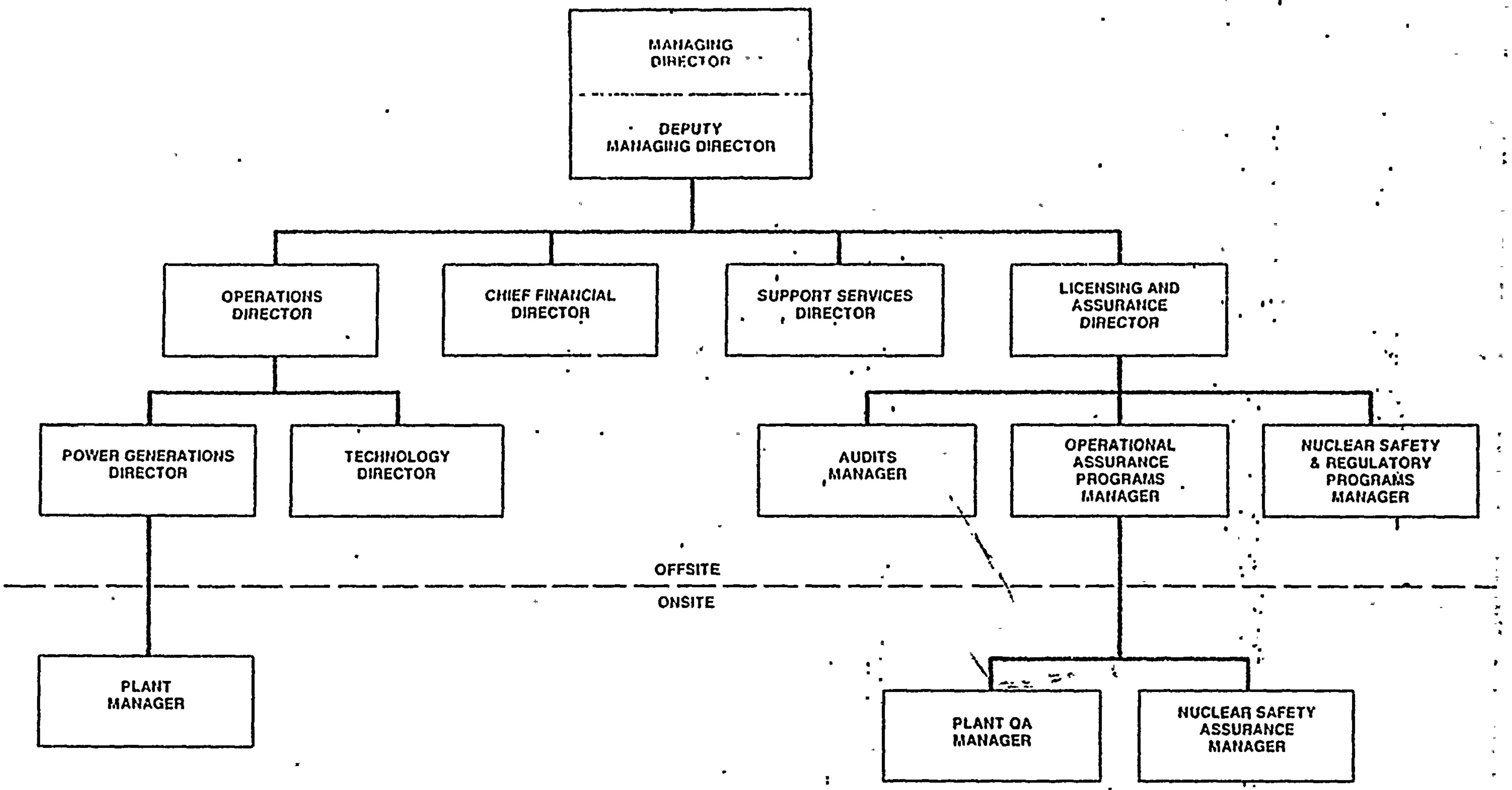

FIGURE 6.2.1-1
OFFSITE ORGANIZATION

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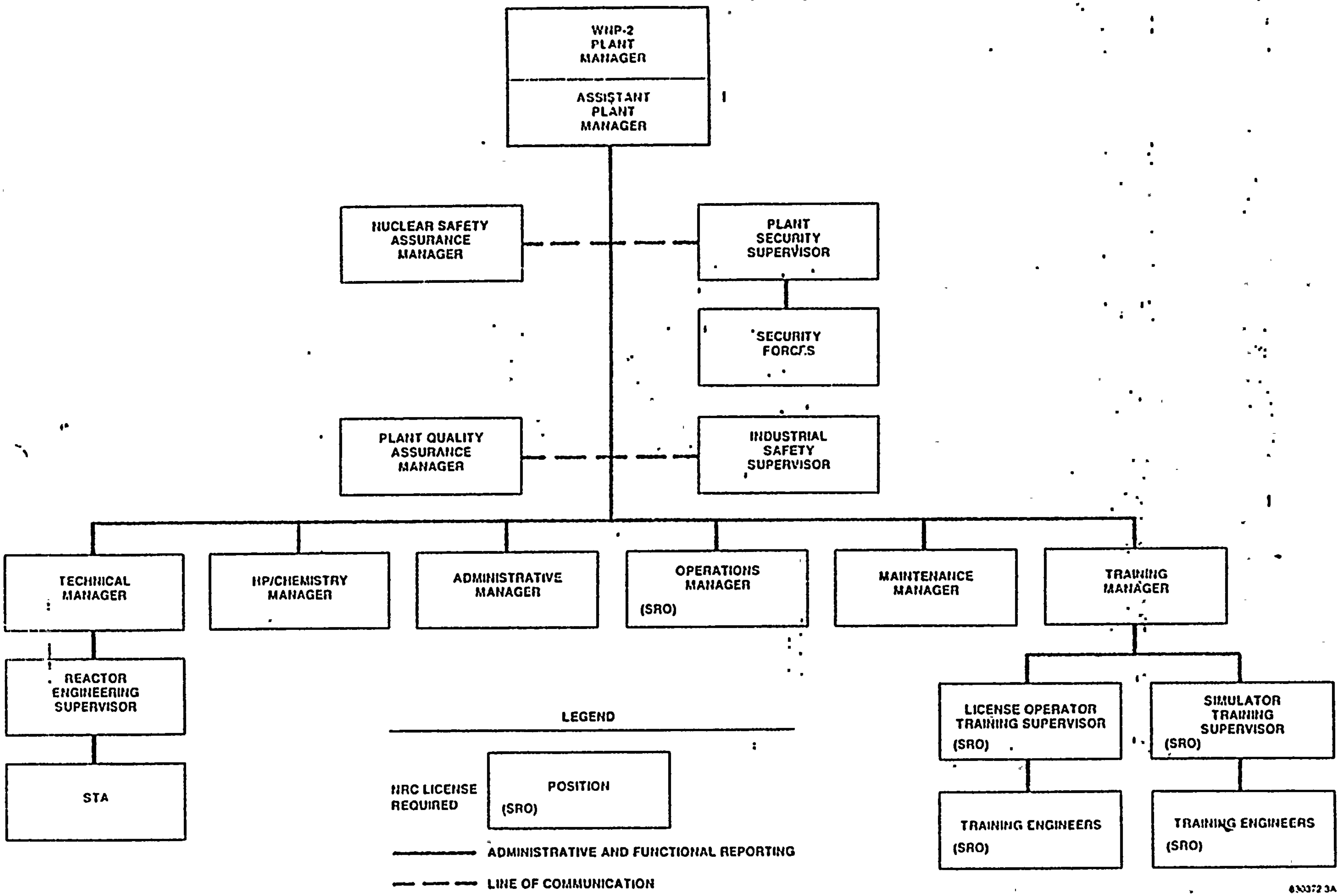
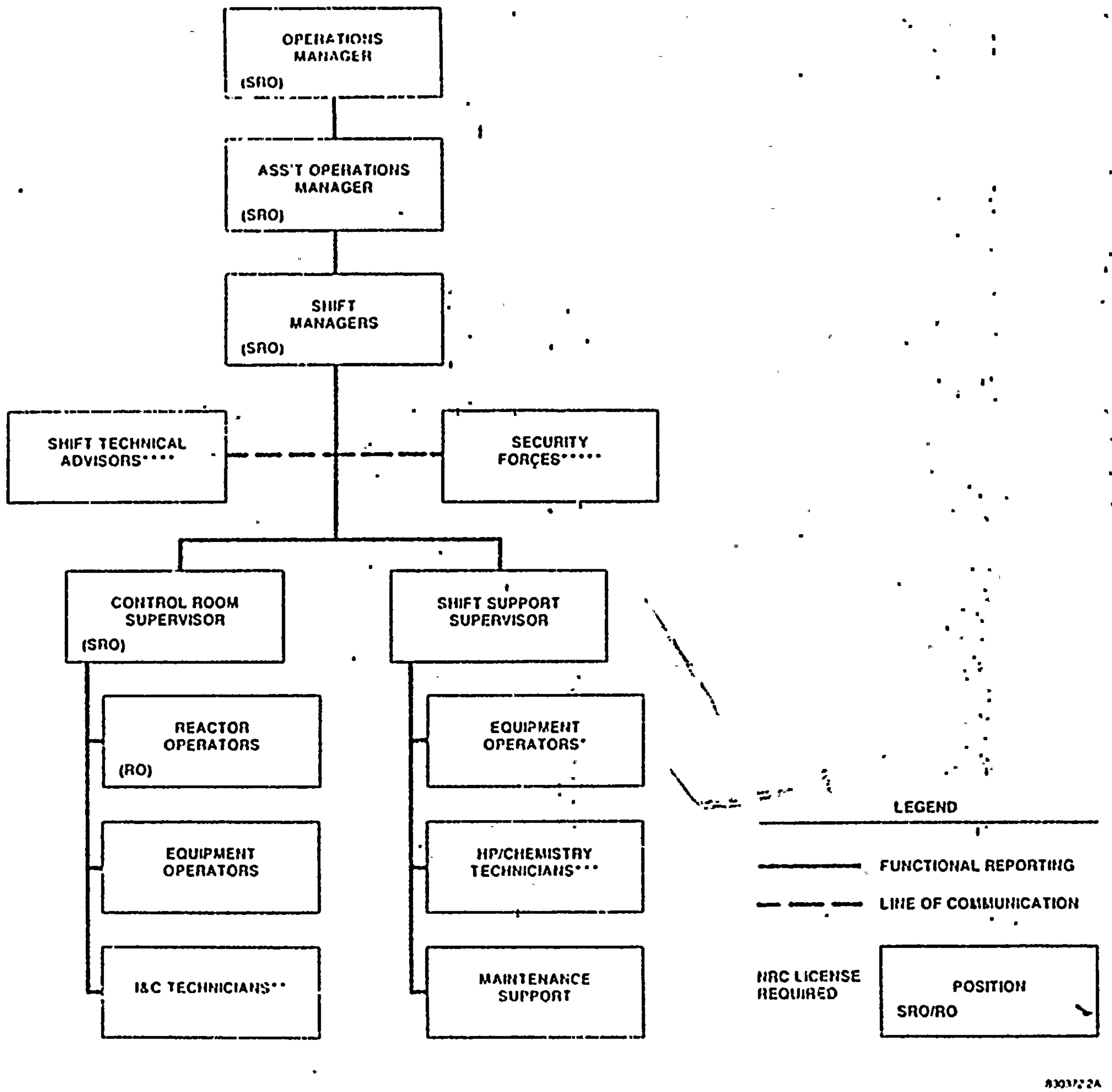


FIGURE 6.2.2-1a
UNIT ORGANIZATION



* TO BE PROVIDED AS NEEDED FROM A POOL OF EQUIPMENT OPERATORS

** TO BE PROVIDED FROM MAINTENANCE DEPARTMENT

*** TO BE PROVIDED FROM HP/CHEM DEPARTMENT

**** TO BE PROVIDED FROM THE TECHNICAL DEPARTMENT AS REQUIRED

***** SEE SECURITY FOR TOTAL SECURITY FORCE STAFFING LEVEL

FIGURE 6.2.2-1b
UNIT ORGANIZATION — OPERATIONS DEPARTMENT

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TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SM	1	1
CRS	1	None
RO	2	1
EO	2	1
STA	1	None

TABLE NOTATION

SM - Shift Manager with a Senior Operator license ~~on Unit 1.~~ for WNP-2
CRS - Individual with a Senior Operator license ~~on Unit 1.~~ for WNP-2
RO - Individual with a Operator license ~~on Unit 1.~~ for WNP-2
EO - Equipment Operator
STA - Shift Technical Advisor

Except for the Shift Manager, the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Manager from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Manager from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.2.3 ~~SAFETY ENGINEERING GROUP (SEG)~~ NUCLEAR SAFETY ASSURANCE GROUP (NSAG)

FUNCTION

6.2.3.1 The ~~SEG~~ NSAG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ~~SEG~~ NSAG shall make detailed recommendations for revised procedures, equipment and modifications, maintenance activities, operations activities, or other means of improving unit safety to the Director of Safety and Security.

COMPOSITION

6.2.3.2 The ~~SEG~~ NSAG shall be composed of at least ~~five~~ three, dedicated, full-time engineers located on site. Each shall have a bachelor's degree or equivalent in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ~~SEG~~ NSAG shall be responsible for ~~maintaining surveillance of~~ assessing unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ~~ISEG~~ NSAG shall be prepared, maintained, and forwarded each calendar month to ~~(a high level corporate official in a technically oriented position who is not in the management chain for power production)~~ the Operational Assurance Programs Manager

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift ~~Supervisor~~ Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI ~~/ANS 3.1-1978~~ N18.1-1971 for comparable positions, except for the Health Physics/ Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, ~~September 1975~~ Rev. 1-R (May, 1977). The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

*Not responsible for sign-off function.

DRAFT

6.4 TRAINING

6.4.1 A retraining and replacement training program for the Manager unit staff shall be maintained under the direction of the Training ~~Supervisor~~, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS COMMITTEE (POC)

FUNCTION

6.5.1.1 The POC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The POC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman:	Assistant Plant Manager
Member:	Operations ~~Superintendent~~ Manager
Member:	Technical ~~Supervisor Superintendent~~ Manager
Member:	Maintenance ~~Supervisor Superintendent~~ Manager
Member:	Administrative Manager
Member:	Plant QA/QC Manager
Member:	Health Physics/Chemistry Manager

ALTERNATES

or Vice Chairman

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

MEETING FREQUENCY

6.5.1.4 The Plant Operations Committee shall meet at least once per calendar month and as convened by the POC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the POC necessary for the performance of the POC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or ~~his designated alternate~~ Vice Chairman and four members including alternates. No more than two alternates shall make up the quorum.

RESPONSIBILITIES

6.5.1.6 The POC shall be responsible for:

a. Review of (1) all proposed procedures required by Specification 6.8 and changes thereto, (2) all proposed programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to ~~Appendix A~~ the Technical Specifications;

Justification: Tech. Spec's no longer segregated into 2 appendices.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Director of Power Generation and to the Corporate Nuclear Safety Review Board;

f. Review of events requiring 24-hour written notification to the Commission;

g. Review of unit operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager or the Corporate Nuclear Safety Review Board;

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Corporate Nuclear Safety Review Board; and

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Corporate Nuclear Safety Review Board.

AUTHORITY

6.5.1.7 The POC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a through d prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a through e constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Director of Power Generation and the Corporate Nuclear Safety Review Board of disagreement between the POC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The POC shall maintain written minutes of each POC meeting that, at a minimum, document the results of all POC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Director of Power Generation and the Corporate Nuclear Safety Review Board.

6.5.2 CORPORATE NUCLEAR SAFETY REVIEW BOARD (CNSRB)

FUNCTION

6.5.2.1 The CNSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The CNSRB shall report to and advise the Managing Director on those areas of responsibility in Specifications 6.5.2.7. ~~and 6.5.2.8~~.

COMPOSITION

6.5.2.2 The CNSRB shall be composed of the:

Chairman:	Nuclear Safety & Regulatory Programs Manager ~~Director of Safety and Security~~
Alternate Chairman:	Director of ~~Quality~~ Assurance Licensing &
Member:	Director of Power Generation
Member:	Director of Technology
Member:	Project 1/4 Director/Plant Manager
Member:	Project 2 Director/Plant Manager
Member:	Project 3/5 Director/Plant Manager
Executive Secretary:	Appointed by Managing Director
Member at Large:	Appointed by Managing Director
Member at Large:	Appointed by Managing Director

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNSRB Chairman to serve on a temporary basis ~~however, no more than two alternates shall participate as voting members in CNSRB activities at any one time.~~ when a regular member will not be available.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNSRB ~~Director~~ Committee to provide expert advice to the CNSRB.

MEETING FREQUENCY

6.5.2.5 The CNSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter. Meetings may be called on a more frequent basis at the discretion of the Chairman.

QUORUM

6.5.2.6 The quorum of the CNSRB necessary for the performance of the CNSRB review and audit functions of these Technical Specifications shall consist of the Chairman or ~~his designated~~ alternate, chairman and at least four CNSRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The CNSRB shall review:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. Events requiring 24-hour written notification to the Commission;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the POC.

j. Audit reports and summary reports of audits (see 6.5.3)

DRAFT

6.5.3 AUDITS

6.5.~~3.2~~ 3.1 Audits of unit activities shall be performed under the cognizance of the ~~[illegible]~~ ~~Manager of Audits~~ Audits Manager. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire plant staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in plant equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per ~~24~~ 12 months per 10CFR50.54(t).

f. The Security Plan and implementing procedures at least once per 24 months.

g. The Fire Protection Program and implementing procedures at least once per 24 months.

h. Any other area of plant operation considered appropriate by the Director of Quality Assurance.

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In addition to the above, the Director of Quality Assurance shall ensure that the following inspections and audits are performed and will be cognizant of the results:

(1) An independent fire protection and loss prevention inspection and audit at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

(2) An inspection and audit of the fire protection and loss prevention by an outside qualified fire consultant at intervals no greater than 36 months.

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6.5.3.2 Records of audit activities shall be prepared, approved and distributed as indicated below:

a. Audit reports encompassed by Specifications 6.5.3.1 above, shall be forwarded to the CNSRB Chairman, the Managing Director, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

b. A summary report of audits encompassed by Specifications 6.5.3.1 above, shall be forwarded to the CNSRB Chairman at least once per 6 months.

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6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Specification 6.9, and

b. Each REPORTABLE OCCURRENCE requiring 24-hour notification to the Commission shall be reviewed by the POC, and the results of this review shall be submitted to the CNSRB and the Director of Power Generation.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Director of Power Generation and the CNSRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the POC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the CNSRB, and the Director of Power Generation within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the POC and shall be approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed by the POC, and approved by the Plant Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the {LPCS, HPCS, RHR, RCIC, hydrogen recombiner, process sampling, containment and the standby gas treatment} systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

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

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

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6.9.1.8 Semiannual Radioactive Effluent Release Report[3/]

a. Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

b. The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water- Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

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c. The radioactive effluent release report shall include the following information for each type of solid waste shipped offsite during the report period:

   a. Container volume,

   b. Total curie quantity (specify whether determined by measurement or estimate),

   c. Principal radionuclides (specify whether determined by measurement or estimate),

   d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

   e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

   f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

3/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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STARTUP REPORT (Continued)

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

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions; and

b. Documentation of all challenges to main steam line safety/relief valves.

N/A per appendix B (item II.K.3.3) of WNP-2 FSAR

MONTHLY OPERATING REPORTS

6.9.1.~~8~~9 Routine reports of operating statistics and shutdown experience ~~(, including documentation of all challenges to the main steam system safety/relief valves,)~~ shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.~~9~~10 The REPORTABLE OCCURRENCES of Specifications 6.9.1.~~10~~11 and 6.9.1.~~11~~12 including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a Licensee Event Report shall be completed and reference shall be made to the original report date.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.11 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office of the NRC or his designee no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the Limiting Safety System Setting in the Technical Specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a Limiting Condition for Operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the Safety Analysis Report.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the Safety Analysis Report.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

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PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP ~~(Continued)~~

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Failure of main steam line safety/relief valves.

THIRTY DAY WRITTEN REPORTS

6.9.1.~~X~~12 The types of events listed below shall be the subject of written reports to the Regional Administrator of the Regional Office of the NRC within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety features instrumentation settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features systems.

d. Abnormal degradation of systems other than those specified in Specification 6.9.1.~~X~~c designed to contain radioactive material resulting from the fission process.

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the life of the item or duration of the unit Operating License, as appropriate;

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

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RECORD RETENTION (Continued)

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Quality assurance records which have been designated as lifetime records ~~Records of quality assurance activities required by the Operational Quality Assurance Manual~~ following the recommendations of ANSI N45.2.9-1974.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the POC and the CNSRB.

l. Records of the service lives of all hydraulic and mechanical snubbers listed on Table~~(s)~~ ~~3.7.5-1~~ ~~(and 3.7.5-2)~~ 3.7.4-1 including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

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HIGH RADIATION AREA (Continued)

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or the health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote (such as use of closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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*Measurement made at 18 inches from source of radioactivity.

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6.~~X~~13 PROCESS CONTROL PROGRAM (PCP)

6.~~X~~13.1 The PCP shall be approved by the Commission prior to implementation.

6.~~X~~13.2 Licensee initiated changes to the PCP:

a. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

6.~~X~~14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.~~X~~14.1 The ODCM shall be approved by the Commission prior to implementation.

6.~~X~~14.2 Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission within 90 days of the date the change(s) was made effective. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.