Docket Nos. 50-282
and 50-306

August 10, 1994

Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Anderson:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INSTRUMENTATION SPECIFICATION CHANGES (TAC NOS. M84671 AND M84672)

The Commission has issued the enclosed Amendment No.111 to Facility Operating License No. DPR-42 and Amendment No. 104 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated September 21, 1992, as revised December 29, 1992, November 24, 1993, May 17, 1994, and June 21, 1994.

The amendments revise Technical Specifications and associated Bases for surveillance test intervals and allowed outage times for the engineered safety features and reactor protection system instrumentation consistent with the NRC staff position as documented in NRC letters to the Westinghouse Owners Group. The amendments also update operation modes to be consistent with Westinghouse Standard Technical Specification operational modes and also include several editorial changes to the Prairie Island Technical Specifications that are unrelated to the changes described above.

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,

Original signed by

Marsha Gamberoni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 111 to DPR-42
2. Amendment No. 104 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

#94-138

OFFICE	LA:PD31	PM:PD31	BC:EELB	BC:SRXB	BC:OTSB	OGC	PD:PD31
NAME	CJamerson	MGamberoni:gll	CBerlinger	TCollins	CGrimes	R Bachmann	LBMarsh
DATE	06/22/94	06/22/94	/ /94	/ /94	7/15/94	7/18/94	8/8/94

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Mr. Roger O. Anderson, Director
Northern States Power Company

Prairie Island Nuclear Generating
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company (the licensee) dated September 21, 1992, as revised December 29, 1992, November 24, 1993, May 17, 1994, and June 21, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 111 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 111

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS-xii	TS-xii
TS.1-1	TS.1-1
TS.1-2	TS.1-2
TS.1-3	TS.1-3
TS.1-4	TS.1-4
TS.1-5	TS.1-5
TS.1-7	TS.1-7
TS.1-8	TS.1-8
--	TABLE TS.1-1
TS.2.3-3	TS.2.3-3
TS.2.3-4	TS.2.3-4
TS.3.5-1	TS.3.5-1
TABLE TS.3.5-2 (Pages 1 & 2)	TABLE TS.3.5-2A (Pages 1 - 6)
--	TABLE TS.3.5-2B (Pages 1 - 9)
TABLE TS.3.5-3 (Pages 1 & 2)	--
TABLE TS.3.5-4 (Pages 1 & 2)	--
TABLE TS.3.5-5	--
TABLE TS.3.5-6	--
TS.3.10-1	TS.3.10-1
TS.3.10-2	TS.3.10-2
TS.4.1-1	TS.4.1-1
TABLE TS.4.1-1 (Pages 1 - 5)	TABLE TS.4.1-1A (Pages 1 - 5)
--	TABLE TS.4.1-1B (Pages 1 - 7)
--	TABLE TS.4.1-1C (Pages 1 - 4)
TABLE TS.4.1-2B (Pages 1 & 2)	TABLE TS.4.1-2B (Pages 1 & 2)
B.2.3-2	B.2.3-2
B.2.3-3	B.2.3-3
B.3.5-1	B.3.5-1
B.3.5-2	B.3.5-2
B.3.5-3	B.3.5-3
B.3.5-4	B.3.5-4
B.3.5-5	B.3.5-5
B.3.10-1	B.3.10-1
B.3.10-2	B.3.10-2
B.4.1-1	B.4.1-1
B.4.1-2	--

TECHNICAL SPECIFICATIONS

LIST OF TABLES

1.0 DEFINITIONS

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

ACTION

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

AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY

AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY shall exist when:

1. Single doors in the Auxiliary Building Special Ventilation Zone are locked closed, and
2. At least one door in each Auxiliary Building Special Ventilation Zone air lock type passage is closed, and
3. The valves and actuation circuits that isolate the Auxiliary Building Normal Ventilation System following an accident are OPERABLE.
4. The Auxiliary Building Special Ventilation System is OPERABLE.

CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

CHANNEL CALIBRATION

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

CHANNEL RESPONSE TEST

A CHANNEL RESPONSE TEST consists of injecting a simulated signal into the channel as near the sensor as practicable to measure the time for electronics and relay actions, including the output scram relay.

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

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

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

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

2. The equipment hatch is closed and sealed.

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

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

CORE ALTERATION

CORE ALTERATION is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel, which may affect core reactivity. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.6. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

FIRE SUPPRESSION WATER SYSTEM

The FIRE SUPPRESSION WATER SYSTEM consists of: Water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

GASEOUS RADWASTE TREATMENT SYSTEM

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

LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are settings, as specified in Section 2.3, for automatic protective devices related to those variables having significant safety functions.

MEMBERS OF THE PUBLIC

MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM is the manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents, in the calculation of liquid and gaseous effluent monitoring instrumentation alarm and/or trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program.

OPERABLE - OPERABILITY

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

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this paragraph.

The OPERABILITY of a system or component shall be considered to be established when: (1) it satisfies the Limiting Conditions for Operation in Specification 3.0, (2) it has been tested periodically in accordance with Specification 4.0 and has met its performance requirements, and (3) its condition is consistent with the two paragraphs above.

OPERATIONAL MODE - MODE

An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table TS.1.1.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental characteristics of the core and related instrumentation. PHYSICS TESTS are conducted such that the core power is sufficiently reduced to allow for the perturbation due to the test and therefore avoid exceeding power distribution limits in Specification 3.10.B.

Low power PHYSICS TESTS are run at reactor powers less than 2% of rated power.

RATED THERMAL POWER

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

REPORTABLE EVENT

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

SHIELD BUILDING INTEGRITY

SHIELD BUILDING INTEGRITY shall exist when:

1. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed, and
2. The shield building equipment opening is closed.
3. The Shield Building Ventilation System is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which:

1) the reactor is subcritical

   or

2) the reactor would be subcritical from its present condition assuming all rod cluster control assemblies are fully inserted except for the rod cluster control assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the specified Surveillance Frequency so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

For example, the surveillance frequency for the automatic trip and interlock logic specifies that the functional testing of that system is monthly and that each train shall be tested at least every two months on a STAGGERED TEST BASIS. Per the definition above, for the automatic trip and interlock logic, the Surveillance Frequency interval is monthly and the number of trains (channels) is 2 (n=2). Therefore, STAGGERED TEST BASIS requires one train be tested each month such that after two Surveillance Frequency intervals (two months) both trains will have been tested.

STARTUP OPERATION

The process of heating up a reactor above 200°F, making it critical, and bringing it up to POWER OPERATION.

THERMAL POWER

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

UNRESTRICTED AREAS

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

TABLE TS.1-1

OPERATIONAL MODES

MODE	TITLE	REACTIVITY CONDITION	%RATED THERMAL POWER	AVERAGE COOLANT TEMPERATURE	REACTOR VESSEL HEAD CLOSURE BOLTS FULLY TENSIONED
1	POWER OPERATION	Critical	> 2%	NA	YES
2	HOT STANDBY**	Critical	≤ 2%	NA	YES
3	HOT SHUTDOWN**	Subcritical	NA	≥ 350°F	YES
4	INTERMEDIATE SHUTDOWN**	Subcritical	NA	< 350°F ≥ 200°F	YES
5	COLD SHUTDOWN	Subcritical	NA	< 200°F	YES
6	REFUELING	NA*	NA	NA	NO

* Boron concentration of the reactor coolant system and the refueling cavity sufficient to ensure that the more restrictive of the following conditions is met:

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

b. Boron concentration ≥ 2000 ppm.

** Prairie Island specific MODE title, not consistent with Standard Technical Specification MODE titles. MODE numbers are consistent with Standard Technical Specification MODE numbers.

2.3.A.2.g. Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

h. Open reactor coolant pump motor breaker.

Reactor coolant pump bus underfrequency - ≥58.2 Hz

i. Power range neutron flux rate.

1. Positive rate - ≤15% of RATED THERMAL POWER with a time constant ≥2 seconds

2. Negative rate - ≤7% of RATED THERMAL POWER with a time constant ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

c. Turbine Generator trip

1. Turbine stop valve indicators - closed

2. Low auto stop oil pressure - ≥45 psig

d. Safety injection - See Specification 3.5

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

1. P-6 Interlock:

   Source range high flux trip shall be unblocked whenever intermediate range neutron flux is $\leq 10^{-10}$ amperes.

2. P-7 Interlock:

   "At power" reactor trips that are blocked at low power (low pressurizer pressure, high pressurizer level, and loss of flow for one or two loops) shall be unblocked whenever:

   a. Power range neutron flux is ≥12% of RATED THERMAL POWER or,

   b. Turbine load is ≥10% of full load turbine impulse pressure.

3. P-8 Interlock:

   Low power block of single loop loss of flow is permitted whenever power range neutron flux is ≤10% of RATED THERMAL POWER.

4. P-9 Interlock:

   Reactor trip on turbine trip shall be unblocked whenever power range neutron flux is ≥50% of RATED THERMAL POWER.

5. P-10 Interlock:

   Power range high flux low setpoint trip and intermediate range high flux trip shall be unblocked whenever power range neutron flux is ≤9% of RATED THERMAL POWER.

C. Control Rod Withdrawal Stops

1. Block automatic rod withdrawal:

   a. P-2 Interlock:

      Turbine load ≤15% of full load turbine impulse pressure.

3.5 INSTRUMENTATION SYSTEM

Applicability

Applies to protection system instrumentation.

Objectives

To provide for automatic initiation of the engineered safety features in the event the principal process variable limits are exceeded, and to delineate the conditions of the reactor trip and engineered safety feature instrumentation necessary to ensure reactor safety.

Specification

A. Limiting set points for instrumentation which initiates operation of the engineered safety features shall be as stated in Table TS.3.5-1.

B. For on-line testing or in the event of failure of a sub-system instrumentation channel, plant operation shall be permitted to continue at RATED THERMAL POWER in accordance with Tables TS.3.5-2A and TS.3.5-2B.

TABLE TS.3.5-2A (Page 1 of 6)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3[a], 4[a], 5[a]	8
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1[b], 2	2
3. Power Range, Neutron Flux, High Positive Rate	4	2	3	1, 2	2
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5. Intermediate Range, Neutron Flux	2	1	2	1[b], 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2[c]	4
b. Shutdown	2	1	2	3[a], 4[a], 5[a]	5
7. Overtemperature ΔT	4	2	3	1, 2	6
8. Overpower ΔT	4	2	3	1, 2	6

(a) When the Reactor Trip Breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

(b) Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(c) Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

TABLE TS.3.5-2A (Page 2 of 6)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Low Pressurizer Pressure	4	2	3	1	6
10. High Pressurizer Pressure	3	2	2	1, 2	6
11. Pressurizer High Water Level	3	2	2	1	6
12. Reactor Coolant Flow Low	3/loop	2/loop	2/loop	1	6
13. Turbine Trip					
a. Low AST Oil Pressure	3	2	2	1	6
b. Turbine Stop Valve Closure	2	2	1	1	6
14. Lo-Lo Steam Generator Water Level	3/SG	2/SG in any SG	2/SG in each SG	1, 2	6
15. Undervoltage on 4.16 kV Buses 11 and 12 (Unit 2: 21 and 22)	2/bus	1/bus on both buses	2 on one bus	1	11

TABLE TS.3.5-2A (Page 3 of 6)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Loss of Reactor Coolant Pump					
a. RCP Breaker Open	1/pump	1	1/pump	1	1
b. Underfrequency 4kV bus	2/bus	1/bus on both buses	2 on one bus	1	11
17. Safety Injection Input from ESF	2	1	2	1, 2	7
18. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3(a), 4(a), 5(a)	7 8
19. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3(a), 4(a), 5(a)	9 8
20. Reactor Trip Bypass Breakers	2	1	1	(d)	10

(a) When the Reactor Trip Breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

(d) When the Reactor Trip Bypass Breakers are racked in and closed for bypassing a Reactor Trip Breaker and the Control Rod System is capable of rod withdrawal.

TABLE 3.5-2A (Page 4 of 6)

Action Statements

ACTION 1: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours.

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

a. The inoperable channel is placed in the tripped condition within 6 hours;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1; and

c. If THERMAL POWER is above 85% of RATED THERMAL POWER, then determine the core quadrant power balance in accordance with the requirements of Specification 3.10.C.4.

d. One additional channel may be taken out of service for low power PHYSICS TESTS.

ACTION 3: With the number of channels OPERABLE one less than the Total Number of Channels and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below the P-10 (Power Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-10 Setpoint.

ACTION 4: With the number of OPERABLE channels one less than the Total Number of Channels suspend all operations involving positive reactivity changes.

ACTION 5: With the number of OPERABLE channels one less than the Total Number of Channels, suspend all operations involving positive reactivity changes, and restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers.

TABLE 3.5-2A (Page 5 of 6)

Action Statements

ACTION 6: With the number of OPERABLE channels one less than the Total Number of Channels, HOT STANDBY and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1.

ACTION 7: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT SHUTDOWN within the next 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 8: With the number of OPERABLE channels one less than the Total Number of Channels restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 9:

a. With one of the diverse trip features (Undervoltage or Shunt Trip Attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply the requirements of b below. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance and testing to restore the diverse trip feature to OPERABLE status.

b. With one of the Reactor Trip Breakers otherwise inoperable, be in at least HOT SHUTDOWN within 6 hours; however, one Reactor Trip Breaker may be bypassed for up to 4 hours for surveillance testing per Specification 4.1, provided the other Reactor Trip Breaker is OPERABLE.

ACTION 10: With the Reactor Trip Bypass Breaker inoperable, restore the Reactor Trip Bypass Breaker to OPERABLE status prior to using the Reactor Trip Bypass Breaker to bypass a Reactor Trip Breaker. If the Reactor Trip Bypass Breaker is racked in and closed for bypassing a Reactor Trip Breaker and it becomes inoperable, be in at least HOT SHUTDOWN within 6 hours. Restore the Bypass Breaker to OPERABLE status within the next 48 hours or open the Bypass Breaker within the following hour.

TABLE 3.5-2A (Page 6 of 6)

Action Statements

ACTION 11: With the number of OPERABLE channels less than the Total Number of Channels, POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel(s) is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel(s) may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1.

ACTION 12: NOT USED

ACTION 13: NOT USED

ACTION 14: NOT USED

ACTION 15: NOT USED

ACTION 16: NOT USED

ACTION 17: NOT USED

ACTION 18: NOT USED

ACTION 19: NOT USED

TABLE TS.3.5-2B (Page 1 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	23
b. High Containment Pressure	3	2	2	1, 2, 3, 4	24
c. Steam Line Low Pressure	3/Loop	2 in any Loop	2/Loop	1, 2, 3[a]	24
d. Pressurizer Low Pressure	3	2	2	1, 2, 3[a]	24
e. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	20
2. CONTAINMENT SPRAY					
a. Manual Initiation	2	2	2	1, 2, 3, 4	23
b. Hi-Hi Containment Pressure	3 channels with 2 sensors per channel	1 sensor per channel in all 3 channels	1 sensor per channel in all 3 channels	1, 2, 3, 4	21
c. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	20

(a) Trip function may be blocked in this MODE below a Reactor Coolant System Pressure of 2000 psig.

TABLE TS.3.5-2B (Page 2 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. CONTAINMENT ISOLATION					
a. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
b. Manual	2	1	2	1, 2, 3, 4	23
c. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	20
4. CONTAINMENT VENTILATION ISOLATION					
a. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
b. Manual	2	1	2	(b)	22
c. Manual Containment Spray	See Functional Unit 2a above for Manual Containment Spray requirements.				
d. Manual Containment Isolation	See Functional Unit 3b above for Manual Containment Isolation requirements.				
e. High Radiation in Exhaust Air	2	1	2	(b)	22
f. Automatic Actuation Logic and Actuation Relays	2	1	2	(b)	22

(b) Whenever CONTAINMENT INTEGRITY is required and either of the containment purge systems are in operation.

TABLE TS.3.5-2B (Page 3 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. STEAM LINE ISOLATION					
a. Manual	1/Loop	1/Loop	1/Loop	1, 2, 3(c)	27
b. Hi-Hi Containment Pressure	3	2	2	1, 2, 3(c)	24
c. Hi-Hi Steam Flow with Safety Injection					
1. Hi-Hi Steam Flow	2/Loop	1 in any Loop	1/Loop	1, 2, 3(c)	29
2. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
d. Hi Steam Flow and 2 of 4 Lo-Lo T_{avg} with Safety Injection:					
1. Hi Steam Flow	2/Loop	1 in any Loop	1/Loop	1, 2, 3(d)	29
2. Lo-Lo T_{avg}	4	2	3	1, 2, 3(d)	24
3. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				

(c) When either main steam isolation valve is open.

(d) When reactor coolant system average temperature is greater than 520°F and either main steam isolation valve is open.

TABLE TS.3.5-2B (Page 4 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. STEAM LINE ISOLATION (continued)					
e. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3[(c)]	25
6. FEEDWATER ISOLATION					
a. Hi-Hi Steam Generator Level	3/SG	2/SG in any SG	2/SG in each SG	1, 2	24
b. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
c. Reactor Trip with 2 of 4 Low T_{avg} (Main Valves only):					
1. Reactor Trip	2	1	2	1, 2	28
2. Low T_{avg}	4	2	3	1, 2	24
d. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	28

(c) When either main steam isolation valve is open.

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. AUXILIARY FEEDWATER					
a. Manual	2	1	2	1, 2, 3	26
b. Steam Generator Lo-Lo Water Level	3/SG	2/SG in any SG	2/SG in each SG	1, 2, 3	24
c. Undervoltage on 4.16 kV Buses 11 and 12 (Unit 2: 21 and 22) (Start Turbine Driven Pump only)	2/bus	1/bus on both buses	2 on one bus	1, 2	29
d. Trip of Both Main Feedwater Pumps					
1. Turbine Driven	2	2	2	1, 2	26
2. Motor Driven	2	2	2	1, 2	26
e. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
f. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	30

TABLE TS.3.5-2B (Page 6 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. LOSS OF POWER					
a. Degraded Voltage 4kV Safeguards Bus	4/Bus (2/phase on 2 phases)	2/Bus (1/phase on 2 phases)	3/Bus	1, 2, 3, 4	31, 32, 33
b. Undervoltage 4kV Safeguards Bus	4/Bus (2/phase on 2 phases)	2/Bus (1/phase on 2 phases)	3/Bus	1, 2, 3, 4	31, 32, 33

TABLE 3.5-2B (Page 7 of 9)

Action Statements

ACTION 20: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 21: With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed provided the inoperable channel(s) is placed in the tripped condition within 6 hours and the Minimum Channels OPERABLE requirement is met. One inoperable channel may be bypassed at a time for up to 4 hours for surveillance testing per Specification 4.1.

ACTION 22: With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 23: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 24: With the number of OPERABLE channels one less than the Total Number of Channels, operation in the applicable MODE may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1.

TABLE 3.5-2B (Page 8 of 9)

Action Statements

ACTION 25: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT SHUTDOWN within the next 6 hours. Operation in HOT SHUTDOWN may proceed provided the main steam isolation valves are closed, if not, be in at least INTERMEDIATE SHUTDOWN within the following 6 hours. However, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 26: With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.4.2.

ACTION 27: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and close the associated valve.

ACTION 28: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT SHUTDOWN within the next 6 hours. However, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 29: With the number of OPERABLE channels less than the Total Number of Channels, operation in the applicable MODE may proceed provided the following conditions are satisfied:

a. The inoperable channel(s) is placed in the tripped condition within 6 hours, and,

b. The Minimum Channels OPERABLE requirement is met; however, one inoperable channel may be bypassed at a time for up to 4 hours for surveillance testing of other channels per Specification 4.1

TABLE 3.5-2B (Page 9 of 9)

Action Statements

ACTION 30: With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.4.2. However, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 31: With the number of OPERABLE channels one less than the Total Number of Channels, operation in the applicable MODE may proceed provided the inoperable channel is placed in the bypassed condition within 6 hours.

ACTION 32: With the number of OPERABLE channels two less than the Total Number of Channels, operation in the applicable MODE may proceed provided the following conditions are satisfied:

a. One inoperable channel is placed in the bypassed condition within 6 hours, and,

b. The other inoperable channel is placed in the tripped condition within 6 hours, and,

c. All of the channels associated with the redundant 4kV Safeguards Bus are OPERABLE.

ACTION 33: If the requirements of ACTIONS 30 or 31 cannot be met within the time specified, or with the number of OPERABLE channels three less than the Total Number of Channels, declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.7.B.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distribution and to the limits on control rod operations.

Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during POWER OPERATION, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

A. Shutdown Margin

1. Reactor Coolant System Average Temperature > 200°F

   The SHUTDOWN MARGIN shall be greater than or equal to the applicable value shown in Figure TS.3.10-1 when in HOT SHUTDOWN and INTERMEDIATE SHUTDOWN.

2. Reactor Coolant System Average Temperature ≤ 200°F

   The SHUTDOWN MARGIN shall be greater than or equal to 1%Δk/k when in COLD SHUTDOWN.

3. With the SHUTDOWN MARGIN less than the applicable limit specified in 3.10.A.1 or 3.10.A.2 above, within 15 minutes initiate boration to restore SHUTDOWN MARGIN to within the applicable limit.

B. Power Distribution Limits

1. At all times, except during low power PHYSICS TESTING, measured hot channel factors, F^N_Q and $F^N_{\Delta H}$, as defined below and in the bases, shall meet the following limits:

   $F^N_Q \times 1.03 \times 1.05 \leq (F_Q^{RTP} / P) \times K(Z)$

   $F^N_{\Delta H} \times 1.04 \leq F_{\Delta H}^{RTP} \times [1+ PFDH(1-P)]$

   where the following definitions apply:

   - F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
   - $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.
   - PFDH is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT.
   - K(Z) is a normalized function that limits $F_Q(z)$ axially as specified in the CORE OPERATING LIMITS REPORT.

3.10.B.1. - Z is the core height location.

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

- F^N_Q or $F^N_{\Delta H}$ is defined as the measured F_Q or $F_{\Delta H}$ respectively, with the smallest margin or greatest excess of limit.

- 1.03 is the engineering hot channel factor, F^E_Q, applied to the measured F^N_Q to account for manufacturing tolerance.

- 1.05 is applied to the measured F^N_Q to account for measurement uncertainty.

- 1.04 is applied to the measured $F^N_{\Delta H}$ to account for measurement uncertainty.

2. Hot channel factors, F^N_Q and $F^N_{\Delta H}$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

   (a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

   (b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux difference was last determined, by 10% or more of RATED THERMAL POWER.

   F^N_Q (equil) shall meet the following limit for the middle axial 80% of the core:

   $$F^N_Q \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \le (F_Q^{RTP} / P) \times K(Z)$$

   where V(Z) is specified in the CORE OPERATING LIMITS REPORT and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip set-point by 1% for each percent that the measured F^N_Q or by the factor specified in the CORE OPERATING LIMITS REPORT for each percent that the measured $F^N_{\Delta H}$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

   (b) If the measured F^N_Q (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

      1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

      2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F^N_Q (equil) x 1.03 x 1.05 x V(Z) exceeds the limit.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification

A. Calibration, testing, and checking of instrumentation channels and testing of logic channels shall be performed as specified in Tables TS.4.1-1A, 4.1-1B and 4.1-1C.

B. Equipment tests shall be conducted as specified in Table TS.4.1-2A.

C. Sampling tests shall be conducted as specified in Table TS.4.1-2B.

D. Whenever the plant condition is such that a system or component is not required to be OPERABLE the surveillance testing associated with that system or component may be discontinued. Discontinued surveillance tests shall be resumed less than one test interval before establishing plant conditions requiring OPERABILITY of the associated system or component, unless such testing is not practicable (i.e., nuclear power range calibration cannot be done prior to reaching POWER OPERATION) in which case the testing will be resumed within 48 hours of attaining the plant condition which permits testing to be accomplished.

TABLE TS.4.1-1A (Page 1 of 5)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R(13)	N.A.	1, 2, 3(1), 4(1), 5(1)
2. Power Range, Neutron Flux					
a) High Setpoint	S	D(5, 7) M(6, 7) Q(7, 8) R(7)	Q(18)	R	1, 2
b) Low Setpoint	S	R(7)	S/U(17)	R	1(3), 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(7)	Q	R	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(7)	Q	R	1, 2
5. Intermediate Range, Neutron Flux	S	R(7)	S/U(4)	R	1(3), 2
6. Source Range, Neutron Flux					
a. Startup	S	R(7)	S/U(4)	R	2(2)
b. Shutdown	S	R(7)	Q(10)	R	3(1), 4(1), 5(1)
7. Overtemperature ΔT	S	R	Q	R	1, 2
8. Overpower ΔT	S	R	Q	R	1, 2

TABLE 4.1-1A (Page 2 of 5)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Low Pressurizer Pressure	S	R	Q	N.A.	1
10. High Pressurizer Pressure	S	R	Q	N.A.	1, 2
11. Pressurizer High Water Level	S	R	Q	N.A.	1
12. Reactor Coolant Flow Low	S	R	Q	N.A.	1
13. Turbine Trip					
a. Low AST Oil Pressure	N.A.	R	S/U[4, 11]	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	S/U[4, 11]	N.A.	1
14. Lo-Lo Steam Generator Water Level	S	R	Q	N.A.	1, 2
15. Undervoltage 4KV RCP Bus	N.A.	R	Q	N.A.	1

TABLE TS.4.1-1A (Page 3 of 5)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
16. Loss of Reactor Coolant Pump					
a. RCP Breaker Open	N.A.	R	S/U[4]	N.A.	1
b. Underfrequency 4KV Bus	N.A.	R	Q	N.A.	1
17. Safety Injection Input	N.A.	N.A.	R	N.A.	1, 2
18. Automatic Trip and Interlock Logic	N.A.	N.A.	M[9]	R	1, 2, 3[1], 4[1], 5[1]
19. Reactor Trip Breakers	N.A.	N.A.	M[9, 12]	R	1, 2, 3[1], 4[1], 5[1]
20. Reactor Trip Bypass Breakers	N.A.	N.A.	M[14]	R[15]	See Note (16)

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
M	Monthly
Q	Quarterly
S/U	Prior to each reactor startup
R	Each Refueling Shutdown
N.A.	Not applicable.

TABLE NOTATION

(1) When the Reactor Trip Breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

(2) Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

(3) Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(4) Prior to each startup following shutdown in excess of two days if not done in previous 30 days.

(5) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%.

(6) Single point comparison of incore to excore for axial off-set above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than 2%.

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

(8) Incore - Excore Calibration, above 75% of RATED THERMAL POWER.

(9) Each train shall be tested at least every two months on a STAGGERED TEST BASIS.

TABLE 4.1-1A (Page 5 of 5)

TABLE NOTATIONS Continued)

TABLE NOTATION (Continued)

(10) Quarterly surveillance in MODES 3, 4 and 5 shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(11) Setpoint verification is not applicable.

(12) The Functional Test shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(13) The Functional Test shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(14) Manually trip the undervoltage trip attachment remotely (i.e., from the protection system racks).

(15) Automatic undervoltage trip.

(16) Whenever the Reactor Trip Bypass Breakers are racked in and closed for bypassing a Reactor Trip Breaker and the Control Rod Drive System is capable of rod withdrawal.

(17) Prior to each startup if not done previous week.

(18) Including quadrant power tilt monitor.

(19) Not Used

TABLE TS.4.1-1B (Page 1 of 7)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION					
a. Manual Initiation	N.A.	N.A.	R[20]	N.A.	1, 2, 3, 4
b. High Containment Pressure	S	R	Q	N.A.	1, 2, 3, 4
c. Steam Line Low Pressure	S	R	Q	N.A.	1, 2, 3[21]
d. Pressurizer Low Pressure	S	R	Q	N.A.	1, 2, 3[21]
e. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M[22]	N.A.	1, 2, 3, 4
2. CONTAINMENT SPRAY					
a. Manual Initiation	N.A.	N.A.	R	N.A.	1, 2, 3, 4
b. Hi-Hi Containment Pressure	S	R	Q	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M[22]	N.A.	1, 2, 3, 4

TABLE TS.4.1-1B (Page 2 of 7)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION					
a. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements				
b. Manual	N.A.	N.A.	R	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(22)	N.A.	1, 2, 3, 4
4. CONTAINMENT VENTILATION ISOLATION					
a. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements				
b. Manual	N.A.	N.A.	R	N.A.	See Note (26)
c. Manual Containment Spray	See Functional Unit 2a above for all Manual Containment Spray Surveillance Requirements				
d. Manual Containment Isolation	See Functional Unit 3b above for all Manual Containment Isolation Surveillance Requirements				
e. High Radiation in Exhaust Air	D(25)	R	M	N.A.	See Note (26)
f. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(22)	N.A.	See Note (26)

TABLE TS.4.1-1B (Page 3 of 7)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. STEAM LINE ISOLATION					
a. Manual	N.A.	N.A.	R	N.A.	1, 2, 3(23)
b. Hi-Hi Containment Pressure	S	R	Q	N.A.	1, 2, 3(23)
c. Hi-Hi Steam Flow with Safety Injection					
1. Hi-Hi Steam Flow	S	R	Q	N.A.	1, 2, 3(23)
2. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements				
d. Hi Steam Flow and 2 of 4 Lo-Lo T_{avg} with Safety Injection					
1. Hi Steam Flow	S	R	Q	N.A.	1, 2, 3(23)
2. Lo-Lo T_{avg}	S	R	Q	N.A.	1, 2, 3(24)
3. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements				
e. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(22)	N.A.	1, 2, 3(23)

TABLE TS.4.1-1B (Page 4 of 7)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. FEEDWATER ISOLATION					
a. Hi-Hi Steam Generator Level	S	R	Q	N.A.	1, 2
b. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements				
c. Reactor Trip with 2 of 4 Low T_{avg} (Main Valves Only)					
1. Reactor Trip	N.A.	N.A.	R	N.A.	1, 2
2. Low T_{avg}	S	R	Q	N.A.	1, 2
d. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(22)	N.A.	1, 2

TABLE TS.4.1-1B (Page 5 of 7)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. AUXILIARY FEEDWATER					
a. Manual	N.A.	N.A.	R	N.A.	1, 2, 3
b. Steam Generator Low-Low Water Level	S	R	Q	N.A.	1, 2, 3
c. Undervoltage on 4.16 kV Buses 11 and 12 (Unit 2: 21 and 22) (Start Turbine Driven Pump only)	N.A.	R	R	N.A.	1, 2
d. Trip of Both Main Feedwater Pumps					
1. Turbine Driven	N.A.	N.A.	R	N.A.	1, 2
2. Motor Driven	N.A.	N.A.	R	N.A.	1, 2
e. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements				
f. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(22)	N.A.	1, 2, 3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. LOSS OF POWER					
a. Degraded Voltage 4kV Safeguards Bus	N.A.	R	M	N.A.	1, 2, 3, 4
b. Undervoltage 4kV Safeguards Bus	N.A.	R	M	N.A.	1, 2, 3, 4

TABLE 4.1-1B (Page 7 of 7)

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
M	Monthly
Q	Quarterly
R	Each Refueling Shutdown
N.A.	Not Applicable

TABLE NOTATION

(20) One manual switch shall be tested at each refueling on a STAGGERED TEST BASIS.

(21) Trip function may be blocked in this MODE below a reactor coolant system pressure of 2000 psig.

(22) Each train shall be tested at least every two months on a STAGGERED TEST BASIS.

(23) When either main steam isolation valve is open.

(24) When reactor coolant system average temperature is greater than 520°F and either main steam isolation valve is open.

(25) See Table 4.17-2.

(26) Whenever CONTAINMENT INTEGRITY is required and either of the containment purge systems are in operation.

(27) Not Used

(28) Not Used

(29) Not Used

TABLE TS.4.1-1C (Page 1 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Control Rod Insertion Monitor	M	R	S/U[30]	N.A.	1, 2
2. Analog Rod Position	S	R	S/U[30]	N.A.	1, 2, 3[31], 4[31], 5[31]
3. Rod Position Deviation Monitor	M	N.A.	S/U[30]	N.A.	1, 2
4. Rod Position Bank Counters	S[32]	N.A.	N.A.	N.A.	1, 2, 3[31] 4[31], 5[31]
5. Charging Flow	S	R	N.A.	N.A.	1, 2, 3, 4
6. Residual Heat Removal Pump Flow	S	R	N.A.	N.A.	4[37], 5[37], 6[37]
7. Boric Acid Tank Level	D	R[33]	M[33]	N.A.	1, 2, 3, 4
8. Refueling Water Storage Tank Level	W	R	M	N.A.	1, 2, 3, 4
9. Volume Control Tank Level	S	R	N.A.	N.A.	1, 2, 3, 4
10. Annulus Pressure (Vacuum Breaker)	N.A.	R	R	N.A.	See Note (39)
11. Auto Load Sequencers	N.A.	N.A.	M	N.A.	1, 2, 3, 4
12. Boric Acid Make-up Flow Channel	N.A.	R	N.A.	N.A.	1, 2, 3, 4

TABLE TS.4.1-1C (Page 2 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Containment Sump A, B and C Level	N.A.	R	R	N.A.	1, 2, 3, 4
14. Accumulator Level and Pressure	S	R	R	N.A.	1, 2, 3, 4
15. Turbine First Stage Pressure	S	R	Q	N.A.	1
16. Emergency Plan Radiation Instruments (35)	M	R	M	N.A.	1, 2, 3, 4, 5, 6
17. Seismic Monitors	R	R	N.A.	N.A.	1, 2, 3, 4, 5, 6
18. Coolant Flow - RTD Bypass Flowmeter	S	R	M	N.A.	1, 2, 3(34)
19. CRDM Cooling Shroud Exhaust Air Temperature	S	N.A.	R	N.A.	1, 2, 3(31), 4(31), 5(31)
20. Reactor Gap Exhaust Air Temperature	S	N.A.	R	N.A.	1, 2, 3, 4
21. Post-Accident Monitoring Instruments (Table TS.3.15-1)(36)	M	R	N.A.	N.A.	1, 2
22. Post-Accident Monitoring Radiation Instruments (Table TS.3.15-2)	D	R	M	N.A.	1, 2

TABLE TS.4.1-1C (Page 3 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
23. Post-Accident Monitoring Reactor Vessel Level Instrumentation (Table TS.3.15-3)	M	R	N.A.	N.A.	1, 2
24. Steam Exclusion Actuation	W	Y	M	N.A.	1, 2, 3
25. Overpressure Mitigation	N.A.	R	R	N.A.	4(38), 5
26. Auxiliary Feedwater Pump Suction Pressure	N.A.	R	R	N.A.	1, 2, 3
27. Auxiliary Feedwater Pump Discharge Pressure	N.A.	R	R	N.A.	1, 2, 3
28. NaOH Caustic Stand Pipe Level	W	R	M	N.A.	1, 2, 3, 4
29. Hydrogen Monitors	S	Q	M	N.A.	1, 2
30. Containment Temperature Monitors	M	R	N.A.	N.A.	1, 2, 3, 4
31. Turbine Overspeed Protection Trip Channel	N.A.	R	M	N.A.	1

TABLE 4.1-1C (Page 4 of 4)

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
W	Weekly
M	Monthly
Q	Quarterly
S/U	Prior to each startup
Y	Yearly
R	Each refueling shutdown
N.A.	Not applicable

TABLE NOTATION

(30) Prior to each startup following shutdown in excess of two days if not done in previous 30 days.

(31) When the reactor trip system breakers are closed and the control rod drive system is capable of rod withdrawal.

(32) Following rod motion in excess of six inches when the computer is out of service.

(33) Transfer logic to Refueling Water Storage Tank.

(34) When either main steam isolation valve is open.

(35) Includes those instruments named in the emergency procedure.

(36) Except for containment hydrogen monitors which are separately specified in this table.

(37) When RHR is in operation.

(38) When the reactor coolant system average temperature is less than 310°F.

(39) Whenever CONTAINMENT INTEGRITY is required.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY
1. RCS Gross Activity Determination	5/week
2. RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3. RCS Radiochemistry $\bar{E}$ determination	1/6 months(1) (when at power)
4. RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5. RCS Radiochemistry (2)	Monthly
6. RCS Tritium Activity	Weekly
7. RCS Chemistry (Cl*,F*, O2)	5/Week
8. RCS Boron Concentration*(3)	2/Week (4)
9. RWST Boron Concentration	Weekly
10. Boric Acid Tanks Boron Concentration	2/Week
11. Caustic Standpipe NaOH Concentration	Monthly
12. Accumulator Boron Concentration	Monthly
13. Spent Fuel Pit Boron Concentration	Monthly/Weekly(7)(8)

* Required at all times.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

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

Notes:

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

2. To determine activity of corrosion products having a half-life greater than 30 minutes.

3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.

4. The maximum interval between analyses shall not exceed 5 days.

5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.

6. The maximum interval between analyses shall not exceed 3 days.

7. The minimum spent fuel pool boron concentration from Specification 3.8.B.1.b shall be verified by chemical analysis weekly while a spent fuel cask containing fuel is located in the spent fuel pool.

8. The spent fuel pool boron concentration shall be verified weekly, by chemical analysis, to be within the limits of Specification 3.8.E.2.a when fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases continued

The overpower and overtemperature protection setpoints include the effects of fuel densification on core safety limits.

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss of coolant flow incident:

a. Low reactor coolant flow

b. Low voltage on pump power supply bus

c. Pump circuit breaker opening (low frequency on pump power supply bus opens pump circuit breaker)

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis (Reference 7). The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation.

The reactor coolant pump bus undervoltage trip is a direct reactor trip (not a reactor coolant pump circuit breaker trip) which protects the core against DNB in the event of a loss of power to the reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis (Reference 7).

The reactor coolant pump breaker reactor trip is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the reactor coolant pump breaker reactor trip is the frequency set point, ≥58.2 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error (Reference 2) and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system (Reference 8).

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Bases continued

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of RATED THERMAL POWER.

The other reactor trips specified in 2.3.A.3. above provide additional protection. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of-load transient.

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip compliments the power range nuclear flux high and low trip to assure that the criteria are met for rod ejection from partial power.

The negative power range rate trip provides protection against DNB for control rod drop accidents. Most rod drop events will cause a sufficiently rapid decrease in power to trip the reactor on the negative power range rate trip signal. Any rod drop events which do not insert enough reactivity to cause a trip are analyzed to ensure that the core does not experience DNB. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

References

1. USAR, Section 14.4.1
2. USAR, Section 14.3
3. USAR, Section 14.6.1
4. USAR, Section 14.4.1
5. USAR, Section 7.4.1.1, 7.2
6. USAR, Section 3.3.2
7. USAR, Section 14.4.8
8. USAR, Section 14.1.10

3.5 INSTRUMENTATION SYSTEM

Bases

Instrumentation has been provided to sense accident conditions and to initiate reactor trip and operation of the Engineered Safety Features (Reference 1). The OPERABILITY of the Reactor Trip System and the Engineered Safety System instrumentation and interlocks ensures that: (1) the associated ACTION and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, (3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analysis.

Specified surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The evaluation of surveillance frequencies and out of service times for the reactor protection and engineered safety feature instrumentation described in WCAP-10271 included the allowance for testing in bypass. The evaluation assumed that the average amount of time the channels within a given trip function would be in bypass for testing was 4 hours.

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss-of-coolant accident (LOCA) is provided by a high containment pressure signal backed up by the low pressurizer pressure signal. These conditions would accompany the depressurization and coolant loss during a LOCA.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the low pressurizer pressure signal and, in case of a break within the containment, by the high containment pressure signal.

The safety injection of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Containment Spray

Containment sprays are also actuated by a high containment pressure signal (Hi-Hi) to reduce containment pressure in the event of a loss-of-coolant or steam line break accident inside the containment.

The containment sprays are actuated at a higher containment pressure (approximately 50% of design containment pressure) than is safety injection (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated on coincidence of high containment pressure sensed by three sets of one-out-of-two containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the environment in the event of a loss-of-coolant accident.

Steam Line Isolation

In the event of a steam line break, the steam line stop valve of the affected line is automatically isolated to prevent continuous, uncontrolled steam release from more than one steam generator. The steam lines are isolated on high containment pressure (Hi-Hi) or high steam line flow in coincidence with low T_{avg} and safety injection or high steam flow (Hi-Hi) in coincidence with safety injection. Adequate protection is afforded for breaks inside or outside the containment even when it is assumed that the steam line check valves do not function properly.

Containment Ventilation Isolation

Valves in the containment purge and inservice purge systems automatically close on receipt of a Safety Injection signal or a high radiation signal. Gaseous and particulate monitors in the exhaust stream or a gaseous monitor in the exhaust stack provide the high radiation signal.

Ventilation System Isolation

In the event of a high energy line rupture outside of containment, redundant isolation dampers in certain ventilation ducts are closed (Reference 4).

Safeguards Bus Voltage

Relays are provided on buses 15, 16, 25, and 26 to detect undervoltage and degraded voltage (the voltage level at which safety related equipment may not operate properly). Relays are not provided on 4 kV safeguards bus 27 to detect undervoltage and degraded voltage since voltage is monitored on the 4 kV source safeguards bus (i.e., bus 25 or bus 26) to which it is

3.5 INSTRUMENTATION SYSTEM

Bases continued

Safeguards Bus Voltage (continued)

connected. Upon receipt of an undervoltage signal the automatic voltage restoring scheme is actuated after a short time delay which prevents actuation during normal transients (such as motor starting) and which allows protective relaying operation during faults. When degraded voltage is sensed, two time delays are actuated. The first time delay is long enough to allow for normal transients. The first time delay annunciates that a sustained degraded voltage condition exists and enables logic which will ensure that voltage and timing are adequate for safety injection loads by automatically performing the following upon receipt of a safety injection signal:

1. Auto start the diesel generator;
2. Separate the bus from the grid;
3. Load the bus onto the diesel generator; and
4. Start the load sequencer (including safety injection loads).

The second longer time delay is used to allow the degraded voltage condition to be corrected by external actions within a time period that will not cause damage to operating equipment. If voltage is not restored within that time period, the logic automatically performs the following:

1. Auto start the diesel generator;
2. Separate the bus from the grid;
3. Load the bus onto the diesel generator; and
4. Start the load sequencer.

Auxiliary Feedwater System Actuation

The following signals automatically start the pumps and open the steam admission control valve to the turbine driven pump of the affected unit:

1. Low-low water level in either steam generator
2. Trip of both main feedwater pumps
3. Safety Injection signal
4. Undervoltage on both 4.16 kV normal buses (turbine driven pump only)

Manual control from both the control room and the Hot Shutdown Panel are also available. The design provides assurance that water can be supplied to the steam generators for decay heat removal when the normal feedwater system is not available.

Underfrequency 4kV Bus

The underfrequency 4kV bus trip does not provide a direct reactor trip signal to the reactor protection system. A reactor coolant pump bus underfrequency signal from both buses provides a trip signal to both reactor coolant pump breakers. Trip of the reactor coolant pump breakers results in a reactor trip. The underfrequency trip protects against postulated flow coastdown events.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Limiting Instrument Setpoints (continued)

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant (Reference 2) or steam line break accidents (Reference 3) as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis (Reference 2).

4. The steam line low pressure signal is lead/lag compensated and its set-point is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis (Reference 3).

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full-load flow at the full load pressure in order to protect against large steam break accidents. The coincident low T_{avg} setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam break (Reference 3).

6. Steam generator low-low water level and 4.16 kV Bus 11 and 12 (21 and 22 in Unit 2) low bus voltage provide initiation signals for the Auxiliary Feedwater System. Selection of these setpoints is discussed in the Bases of Section 2.3 of the Technical Specification.

7. High radiation signals providing input to the Containment Ventilation Isolation circuitry are set in accordance with the Radioactive Effluent Technical Specifications. The setpoints are established to prevent exceeding the limits of 10 CFR Part 20 at the SITE BOUNDARY.

8. The degraded voltage protection setpoint is ≥94.8% and ≤96.2% of nominal 4160 V bus voltage. Testing and analysis have shown that all safeguards loads will operate properly at or above the minimum degraded voltage setpoint. The maximum degraded voltage setpoint is chosen to prevent unnecessary actuation of the voltage restoring scheme at the minimum expected grid voltage. The first degraded voltage time delay of 8 ± 0.5 seconds has been shown by testing and analysis to be long enough to allow for normal transients (i.e., motor starting and fault clearing). It is also longer than the time required to start the safety injection pump at minimum voltage. The second degraded voltage time delay is provided to allow the degraded voltage condition to be corrected within a time frame which will not cause damage to permanently connected Class 1E loads.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Limiting Instrument Setpoints (continued)

The undervoltage setpoint is 75 ± 2.5% of nominal bus voltage. The minimum setpoint ensures equipment operates above the limiting value of 75% (of 4000 V) for one minute operation. The 75% maximum setpoint is chosen to prevent unnecessary actuation of the voltage restoring scheme during voltage dips which occur during motor starting. The undervoltage time delay of 4 ± 1.5 seconds has been shown by testing and analysis to be long enough to allow for normal transients and short enough to operate prior to the degraded voltage logic, providing a rapid transfer to an alternate source.

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for CHANNEL CALIBRATION and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in a concurrent channel.

References

1. USAR, Section 7.4.2
2. USAR, Section 14.6.1
3. USAR, Section 14.5.5
4. FSAR, Appendix I

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonymous.

A. Shutdown Margin

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.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, reactor coolant system boron concentration and reactor coolant average temperature. The most restrictive condition occurs at end of life and is associated with a postulated steam line break accident and resulting uncontrolled reactor coolant system cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN (shown in Figure TS.3.10-1 as a function of equilibrium hot full power boron concentration) is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirements are based upon this limiting condition and are consistent with plant safety analysis assumptions. With reactor coolant system average temperature less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

In POWER OPERATION and HOT STANDBY, with $k_{eff} \geq 1$, SHUTDOWN MARGIN is ensured by complying with the rod insertion limitations in Specification 3.10.D. In HOT SHUTDOWN, INTERMEDIATE SHUTDOWN and COLD SHUTDOWN, the SHUTDOWN MARGIN requirements in Specification 3.10.A are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. For REFUELING, the shutdown reactivity requirements are specified in Table TS.1-1.

When in POWER OPERATION and HOT STANDBY, SHUTDOWN MARGIN is determined assuming the fuel and moderator temperatures are at the nominal zero power design temperature of 547°F.

With any rod cluster control assembly not capable of being fully inserted, the reactivity worth of the rod cluster control assembly must be accounted for in the determination of SHUTDOWN MARGIN.

B. Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core of greater than or equal to 1.30 for Exxon fuel and 1.17 for Westinghouse fuel during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. The ECCS analysis was performed in accordance with SECY 83-472. One calculation at the 95% probability level was performed as well as one calculation with

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

B. Power Distribution Control (continued)

all the required features of 10 CFR Part 50, Appendix K. The 95% probability level calculation used the peak linear heat generation rate specified in the CORE OPERATING LIMITS REPORT. The Appendix K calculation used the peak linear heat generation rate specified in the CORE OPERATING LIMITS REPORT for the F_Q limit specified in the CORE OPERATING LIMITS REPORT. Maintaining 1) peaking factors below the F_Q limit specified in the CORE OPERATING LIMITS REPORT during all Condition I events and 2) the peak linear heat generation rate below the value specified in the CORE OPERATING LIMITS REPORT at the 95% probability level assures compliance with the ECCS analysis.

During operation, the plant staff compares the measured hot channel factors, F^N_Q and $F^N_{\Delta H}$, (described later) to the limits determined in the transient and LOCA analyses. The terms on the right side of the equations in Section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

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

The K(Z) function specified in the CORE OPERATING LIMITS REPORT is a normalized function that limits F_Q axially. The K(Z) value is based on large and small break LOCA analyses.

V(Z) is an axially dependent function applied to the equilibrium measured F^N_Q to bound F^N_Q's that could be measured at non-equilibrium conditions. This function is based on power distribution control analyses that evaluated the effect of burnable poisons, rod position, axial effects, and xenon worth.

F^E_Q, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

The 1.05 multiplier accounts for uncertainties associated with measurement of the power distribution with the movable incore detectors and the use of those measurements to establish the assembly local power distribution.

F^N_Q (equil) is the measured limiting F^N_Q obtained at equilibrium conditions during target flux determination.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

4.1 OPERATIONAL SAFETY REVIEW

Bases

CHANNEL CHECK

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequencies set forth are deemed adequate for reactor and steam system instrumentation.

CHANNEL CALIBRATION

Calibration is performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels daily calibration against a thermal power calculation will account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

CHANNEL FUNCTIONAL TESTS

The specified surveillance intervals for the Reactor Protection and Engineered Safety Features instrumentation have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Surveillance intervals were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

CHANNEL RESPONSE TESTS

Measurement of response times for protection channels are performed to assure response times within those assumed for accident analysis (USAR, Section 14).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company (the licensee) dated September 21, 1992, as revised December 29, 1992, November 24, 1993, May 17, 1994, and June 21, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 104, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS-xii	TS-xii
TS.1-1	TS.1-1
TS.1-2	TS.1-2
TS.1-3	TS.1-3
TS.1-4	TS.1-4
TS.1-5	TS.1-5
TS.1-7	TS.1-7
TS.1-8	TS.1-8
--	TABLE TS.1-1
TS.2.3-3	TS.2.3-3
TS.2.3-4	TS.2.3-4
TS.3.5-1	TS.3.5-1
TABLE TS.3.5-2 (Pages 1 & 2)	TABLE TS.3.5-2A (Pages 1 - 6)
--	TABLE TS.3.5-2B (Pages 1 - 9)
TABLE TS.3.5-3 (Pages 1 & 2)	--
TABLE TS.3.5-4 (Pages 1 & 2)	--
TABLE TS.3.5-5	--
TABLE TS.3.5-6	--
TS.3.10-1	TS.3.10-1
TS.3.10-2	TS.3.10-2
TS.4.1-1	TS.4.1-1
TABLE TS.4.1-1 (Pages 1 - 5)	TABLE TS.4.1-1A (Pages 1 - 5)
--	TABLE TS.4.1-1B (Pages 1 - 7)
--	TABLE TS.4.1-1C (Pages 1 - 4)
TABLE TS.4.1-2B (Pages 1 & 2)	TABLE TS.4.1-2B (Pages 1 & 2)
B.2.3-2	B.2.3-2
B.2.3-3	B.2.3-3
B.3.5-1	B.3.5-1
B.3.5-2	B.3.5-2
B.3.5-3	B.3.5-3
B.3.5-4	B.3.5-4
B.3.5-5	B.3.5-5
B.3.10-1	B.3.10-1
B.3.10-2	B.3.10-2
B.4.1-1	B.4.1-1
B.4.1-2	--

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 111 AND 104 TO

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated September 21, 1992, as revised December 29, 1992, November 24, 1993, May 17, 1994, and June 21, 1994, the Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The May 17 and June 21, 1994, letters contained clarifying information and updated TS pages which were within the scope of the initial Federal Register notice. The amendments revise TS and associated Bases for surveillance test intervals and allowed outage times for the engineered safety features and reactor protection system (RPS) instrumentation consistent with the NRC staff position as documented in NRC letters to the Westinghouse Owners Group. The amendments also update operation modes to be consistent with Westinghouse Standard Technical Specification (STS) operational modes and also include several editorial changes to the Prairie Island TS that are unrelated to the changes described above.

2.0 EVALUATION

The proposed changes to the TS and the staff's evaluation of these changes are divided into the following main categories:

2.1 Definitions - TS Section 1.0, 3.10.A and Bases for Section B.3.10

The licensee proposed adding new definitions for the terms ACTION, OPERATIONAL MODE - MODE, STAGGERED TEST BASIS and a new TS Table 1-1 which would replace the existing definition for COLD SHUTDOWN, HOT SHUTDOWN, POWER OPERATION and REFUELING. The licensee deleted the definition of DEGREE OF INSTRUMENTATION REDUNDANCY as this term is no longer used in the Prairie Island TS. The proposed definitions are consistent with the Westinghouse STS Revision 4a or with the improved Westinghouse STS except for the following minor exceptions:

1) The title for MODES 2, 3, and 4 are not consistent with the STS. However, the mode numbers are consistent with the STS. This exception

was needed because of the use of these titles throughout the current Prairie Island TS and plant procedures. Because this is only a minor terminology deviation, the exception is acceptable to the staff.

2) The RATED THERMAL POWER conditions for MODES 1 and 2 are based on 2% of full power rather than 5% as stated in the STS. This exception was needed because current Prairie Island TS and procedures use 2% rated thermal power for Modes 1 and 2. Since a value of 2% compared to 5% is conservative, changing the rated thermal power to 5% would provide no improvement in plant safety and may increase the possibility of operator error as the operators would need to adjust to a new power level value for these operating matters. The staff, therefore, finds the exception acceptable.

3) The status of the reactor vessel head closure bolts is specified in a separate column rather than as an asterisked statement. This is an editorial change and is acceptable to the staff.

4) The reactivity conditions are specified by the terms "critical" and "subcritical" instead of the K_{eff} values which are used in the STS. This exception is required because in the current Prairie Island TS, all operating modes are specified in terms of the shutdown margins. The term "critical" and "subcritical" are consistent with the K_{eff} conditions specified in the STS and, therefore, the staff finds the proposed exception acceptable.

5) The reactivity conditions for refueling are as previously described in the Prairie Island TS, therefore, the staff finds this note acceptable.

The licensee also revised the definition of SHUTDOWN MARGIN in Section 1.0 and the requirements of Shutdown Margin in Section 3.10.A. These are consistent with the STS except that the part of the definition which provided guidance on how to calculate Shutdown Margin was added to the bases for Section 3.10.A, rather than being incorporated directly in Section 1.0. These changes expand the Shutdown Margin requirements to the Intermediate and Cold Shutdown conditions and have no effect on the actual shutdown margin limits in the TS. Therefore, the proposed change is acceptable to the staff.

The staff finds the definitions as proposed in Section 2.1 above, including the exceptions noted, acceptable.

2.2 Clarification to TS Section 2.3.A.2.g and Bases for TS Section B.2.3.

TS Section 2.3.A.2.g and the associated bases for this section were revised to clarify that the reactor coolant pump bus undervoltage reactor trip is the direct undervoltage trip and not the indirect trip which results from the reactor coolant pump circuit breaker undervoltage trip. This change was required because the accident in the Updated Safety Analysis Report (USAR), Section 14.4.8.1, utilizes the direct reactor coolant pump bus undervoltage trip. This change is acceptable to the staff as it is consistent with the assumption of the USAR.

2.3 Editorial Changes to TS Sections 2.3.B.1 through 2.3.B.5, TS 4.1-2B and TS 3.10-1

The licensee proposed the following editorial changes to the TS:

1) TS 2.3.B.1 through 2.3.B.5 were revised to include the headings of the interlock names for clarity.

2) TS Table 4.1-2B was revised to delete the footnote at the bottom of page 2 of 2 and a footnote was added at the bottom of page 1 of 2. The footnote on page 1 of 2 provides the information which was previously provided in the footnote on page 2 of 2. Incorporating the footnote on page 1 of 2 makes the existing requirement more readily apparent to the operators.

3) The term "POWER OPERATION" in TS 3.10-1 has been fully capitalized because it is a defined term. This is consistent with the treatment of other defined terms in the TS.

The staff finds the above changes acceptable, as they are editorial in nature, and will aid the operator in reading the TS.

2.4 Changes to TS Section 3.5 and TS Tables 3.5-2 through 3.5-6

2.4.1 Functional Unit 15 of TS Table 3.5-2

The licensee proposed to delete Functional Unit 15, "Control Rod Misalignment Monitor," from TS Table 3.5-2 as this function is not associated with the RPS. Also, TS 3.10.I specifically defines the actions to be taken if control rod position deviations are identified, or quadrant power tilt monitors (used for control rod misalignment monitoring) are determined to be inoperable. Because the TSs adequately address operability of Functional Unit 15, the staff finds the proposed changes acceptable.

2.4.2 Functional Unit 4 of TS Table 3.5-4

The licensee proposed to delete Functional Unit 4, "Emergency Cooldown Equipment Room Isolation," of TS Table 3.5-4 since this requirement is adequately covered in current TS Section 3.4.C. This change increases the time the instrument channel may be inoperable and makes the action consistent with the actuation logic and actuated component for this function. In accordance with the current TS, if the sensors are inoperable, the plant must be taken to hot shutdown and subsequently to cold shutdown if the minimum operability conditions are not met in 24 hours. If the logic or actuated components (dampers) are inoperable, then the only action required by the current TS is to close the associated damper with no mode change required. Since there is only one sensor per system train, the proposed change is justified as it will require the same action for inoperability of the sensor, actuation logic and actuated components. Based on the above, the staff finds the proposed change acceptable.

2.4.3 Deletion of TS Table 3.5-5

The licensee proposed to delete TS Table 3.5-5, "Instrument Operating Conditions for Ventilation Systems," because these items are adequately covered in TS Sections 3.6F and 3.6G for the ventilation systems themselves. This change does not represent a change in requirements and is considered an editorial change. Therefore, the staff finds the proposed change acceptable.

2.4.4 Functional Unit 10 of TS Table 3.5-2

The licensee proposed to remove the Functional Unit 10 single loop and two loop loss of reactor coolant flow reactor trip requirements which were listed separately. The revised TS listed these trip requirements as a loss of reactor coolant flow reactor trip in Functional Unit 12 of new TS Tables 3.5-2A, and 4.1-1A as a single item with no reference to single loop or two loop trips. The reason for this change is that the P-7 and P-8 interlock which enables the single loop and two loop loss of flow trips have the same setpoint (>10% power). Also, single loop and two loop loss of flow trips utilize the same flow instrumentation. Therefore, there is no need to list both trips separately. Based on the above, the staff considers the proposed change editorial in nature and finds it acceptable.

2.4.5 Changes Based on the WCAP-10271

The following changes to TS Section 3.5 and 4.1 are proposed by the licensee:

1) The licensee replaced TS Table 3.5-2 with new TS Table 3.5-2A. The new table is consistent with the format and content of Westinghouse STS Revision 4a, and also incorporates allowed outage times (AOTs) which were approved by the staff in the safety evaluation issued for Westinghouse Topical Report WCAP-10271.

2) The licensee replaced existing TS Tables 3.5-3, 3.5-4, and 3.5-6 with new TS Table 3.5-2B. This new table is consistent with the format and content of Westinghouse STS Revision 4a, and also incorporates the AOTs which were approved by the staff in the safety evaluation issued for Westinghouse Topical Report WCAP-10271.

3) TS Section 3.5 was revised to refer to new TS Tables 3.5-2A and 3.5-2B. Parts C and D of TS Section 3.5 were replaced by incorporating actions or notes into the new tables as appropriate.

4) The licensee replaced existing TS Table 4.1-1 with new TS Tables 4.1-1A, 4.1-1B, and 4.1-1C for Reactor Trip, Engineered Safety Features, and miscellaneous instrumentation surveillance requirements, respectively. The new tables are consistent with the format and content of Westinghouse STS Revision 4a or with the improved Westinghouse STS, and also incorporate the surveillance frequencies which were approved by the staff in the safety evaluation issued for Westinghouse Topical Report WCAP-10271.

5) TS Section 4.1.A was revised to refer to new Tables 4.1-1A through 4.1-1C. Also, TS Section 4.1.D was revised to delete the sentence concerning the requirement that asterisked items in the tables are required to be operable at all times. This requirement was incorporated into the individual new tables.

The above proposed changes are based on Westinghouse Topical Reports WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated February 21, 1985, Supplement 1-P-A dated May 1986 for the RPS instrumentation, and WCAP-10271-P-A Supplement 2, Revision 1, dated May 1989 for engineered safety features actuation system (ESFAS) instrumentation. The staff previously reviewed this topical report and its supplements and issued safety evaluations dated February 21, 1985, for the RPS, and February 22, 1989, and April 30, 1990, for the ESFAS, which approved changes in surveillance test intervals (STIs) and AOTs for these systems subject to certain plant-specific conditions. The licensee responded to these conditions and provided appropriate justification that Prairie Island instrumentation either meets the requirements specifically stated in the staff's safety evaluation for the above topical reports or the requirements do not apply. The licensee proposed to change STIs and AOTs for the RPS and ESFAS instrumentation in accordance with the topical reports and revised the format and content in accordance with Westinghouse STS Revision 4a or with the improved Westinghouse STS. No other changes were made unless discussed in this safety evaluation. Based on the above, the staff finds the proposed changes to the TS to be acceptable.

2.4.6 Changes to TS Requirements for Auxiliary Feedwater (AFW) System

2.4.6.1 Addition of Surveillance Requirements for AFW System Instrumentation

The licensee added surveillance requirements for AFW system instrumentation as Functional Unit 7 in TS Table 4.1-1B which are consistent with the format and content of the Westinghouse STS Revision 4a. The TS requirement is the same as the current requirement in TS Section 4.8. This change is editorial in nature, and, therefore, the staff finds the proposed change acceptable.

2.4.6.2 Changes to AOT for AFW System

The licensee proposed to increase the AOT for the manual AFW system actuation circuitry and automatic AFW actuation on main feedwater (MFW) pump trip circuitry from 48 hours to 72 hours (Actions 26 and 34 in TS Table 3.5.2.B of the November 24, 1993, submittal). The licensee also proposed to increase the AOT for the AFW pump actuation logic and actuation relays from 6 hours to 72 hours (Action 30 in TS Table 3.5.2.B). The licensee proposed these changes for consistency on the basis that 72 hours out-of- service time is allowed for the AFW pump by TS Section 3.4.B.2 and because the failure of one channel will affect automatic start of only one AFW pump.

During the meeting with the staff on March 1, 1994, and in subsequent discussions with the licensee, the staff asked the licensee to revise the action statement for the AFW system circuitry to state that inoperability of this instrumentation results in the inoperability of an AFW pump and would safeguard buses. These functions initiate a reactor trip signal with a one-out-of-two taken twice logic instead of one-out-of-one logic. In addition to the logic change, the licensee proposed to add Action Statements 31, 32, and 33 to define the AOT in accordance with the Westinghouse STS Revision 4a. Based on the above, the staff finds the proposed changes to the TS to be consistent with the previously approved instrumentation design, and therefore, acceptable.

Several TS pages have been updated to reflect changes made by Amendments that were issued since the initial submittal.

Based on its review of the licensee's submittals, the staff concludes that the proposed changes to the Prairie Island TS for the RPS, ESFAS, and other miscellaneous instrumentation systems are consistent with the current Westinghouse STS Revision 4a and previously approved topical reports and license amendments, are clarifications, or are editorial in nature. The staff, therefore, finds the proposed TS changes to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (59 FR 10012). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Hukam Garg

Date: August 10, 1994

DATED: August 10, 1994

AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File
NRC & Local PDRs
PDIII-1 Reading
J. Roe
J. Zwolinski
L. B. Marsh
C. Jamerson
M. Gamberoni (2)
OGC-WF
D. Hagan, 3206 MNBB
G. Hill (4), P1-22
C. Grimes, 11/F/23
H. Garg, 8/H/3
H. Richings, 8/E/23
S. Saba, 7/E/4
C. Schulten, 11/E/22
ACRS (10)
OPA
OC/LFDCB
W. Kropp, R-III
SEDB

cc: Plant Service list