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

May 31, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
Nine Mile Point Nuclear Station, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: ADOPTION OF TECHNICAL SPECIFICATION TASK FORCE TRAVELER 425, REVISION 3) (CAC NO. MF6061)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment No. 222 to Renewed Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit 1 (NMP1). The amendment consists of changes to the technical specifications (TSs) in response to your application dated May 12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15134A232), as supplemented by letters dated October 22, and November 17, 2015 (ADAMS Accession Nos. ML15295A396 and ML15321A253, respectively). The licensee requested to revise the NMP1 TSs by relocating specific surveillance requirement (SR) frequencies to a licensee-controlled program. In accordance with Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" (ADAMS Accession No. ML071360456).

The amendment approves the adoption of NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specifications Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed TSTF] Initiative 5b" (ADAMS Accession No. ML090850642). The *Federal Register* (FR) notice published on July 6, 2009 (74 FR 31996), announced the availability of TSTF-425, Revision 3.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Brenda Mozafari

Brenda L. Mozafari, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures:
1. Amendment No. 222 to DPR-63
2. Safety Evaluation

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

NINE MILE POINT NUCLEAR STATION, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
Renewed License No. DPR-63

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated May 12, 2015, as supplemented by letters dated October 22, and November 17, 2015, 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 Renewed Facility Operating License No. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, which is attached hereto, as revised through Amendment No. 222, is hereby incorporated into this license. Exelon Generation Company, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License and Technical Specifications

Date of Issuance: May 31, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 222

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
Page 3	Page 3

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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Pg 355b	Pg 355b

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

secondary containment penetration flow path not isolated.

(e) If Specification 3.1.1a(1)(a) is not met while in the refueling condition, then:

Immediately suspend core alterations, except for fuel assembly removal, <u>and</u>

Immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

(2) Reactivity margin - stuck control rods

Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. Inoperable control rods shall be valved out of service, in such positions that Specification 3.1.1a(1)(a) is met. In no case shall the number of non-fully inserted rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the cold shutdown condition. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

(2) Reactivity margin - stuck control rods

Each withdrawn control rod shall be exercised in accordance with the Surveillance Frequency Control Program after the control rod has been withdrawn and power level is greater than the low power set point of the RWM. Insert each withdrawn control rod at least one notch.

This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

LIMITING CONDITION FOR OPERATION

c. Scram Insertion Times

(1) The average scram insertion time of all operable control rods, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.00
90	5.00

(2) Except as noted in 3.1.1.c(3), the maximum insertion scram time, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Maximum Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.12
90	5.30

SURVEILLANCE REQUIREMENT

c. Scram Insertion Times

The maximum scram insertion time shall be demonstrated through measurement for*:

(1) All control rods prior to thermal power exceeding 40% power with reactor pressure above 800 psig, after each major refueling outage or after a reactor shutdown that is greater than 120 days.

(2) Specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods with reactor pressure above 800 psig.

(3) At least 20 control rods, on a rotating basis, on a frequency in accordance with the Surveillance Frequency Control Program, with reactor pressure above 800 psig.

* For single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) Control rods with longer scram insertion time will be permitted provided that no other control rod in a nine-rod square array around this rod has a: (a) Scram insertion time greater than the maximum allowed, (b) Malfunctioned accumulator, (c) Valved out of service in a non-fully inserted position.	
d. Control Rod Accumulators At all reactor operating pressures, a rod accumulator may be out of service provided that no other control rod in a nine-rod square array around this rod has a: (1) Malfunctioned accumulator, (2) Valved out of service in a non-fully inserted position, (3) Scram insertion greater than maximum permissible insertion time.	d. Control Rod Accumulators In accordance with the Surveillance Frequency Control Program check the status of the accumulator pressure and level alarms in the control room.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
If a control rod with a malfunctioned accumulator is inserted "full-in" and valved out of service, it shall not be considered to have a malfunctioned accumulator.	
e. Scram Discharge Volume With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore at least one vent and one drain valve to OPERABLE status within 8 hours.	e. Scram Discharge Volume (SDV) Scram Discharge Volume Vent and Drain Valves shall be demonstrated OPERABLE during Power Operations by: 1. In accordance with the Surveillance Frequency Control Program verifying each valve to be open;* 2. In accordance with the Surveillance Frequency Control Program cycling each valve through at least one complete cycle of full travel; and The Scram Discharge Volume Drain and Vent valves shall be demonstrated OPERABLE In accordance with the Surveillance Frequency Control Program by verifying that: 1. Valves close within 10 seconds after receipt of a signal for control rods to scram; 2. Valves open when the scram signal is reset; 3. Level instrumentation response proves that no blockage in the system exists. * These valves may be closed intermittently for testing under administrative controls.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.2 LIQUID POISON SYSTEM

Applicability:

Applies to the operating status of the liquid poison system.

Objective:

To assure the capability of the liquid poison system to function as an independent reactivity control mechanism and as a post-LOCA suppression pool pH control mechanism.

Specification:

a. During power operating conditions, and whenever the reactor coolant system temperature is greater than 212°F except for reactor vessel hydrostatic or leakage testing with the reactor not critical, the liquid poison system shall be operable except as specified in 3.1.2.b.

b. If a redundant component becomes inoperable, Specification 3.1.2.a shall be considered fulfilled, provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

4.1.2 LIQUID POISON SYSTEM

Applicability:

Applies to the periodic testing requirements for the liquid poison system.

Objective:

To specify the tests required to assure the capability of the liquid poison system for controlling core reactivity.

Specification:

The liquid poison system surveillance shall be performed as indicated below:

a. Overall System Test:

(1) In accordance with the Surveillance Frequency Control Program -

Manually initiate the system from the control room. Demineralized water shall be pumped to the reactor vessel to verify minimum flow rates and demonstrate that valves and nozzles are not clogged.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The liquid poison tank shall contain a minimum of 1325 gallons of boron bearing solution. The solution shall have a sufficient concentration of sodium pentaborate enriched with Boron-10 isotope to satisfy the equivalency equation.

$$\frac{C}{13\%\ wt} \times \frac{628300}{M} \times \frac{Q}{86\ GPM} \times \frac{E}{19.8\%\ Atom} \geq 1$$

Where:

C = Sodium Pentaborate Solution Concentration (Wt %)

M = Mass of Water in Reactor Vessel and Recirculation piping at Hot Rated Conditions (501500 lb)

Q = Liquid Poison Pump Flow Rate (30 GPM nominal)

E = Boron-10 Enrichment (Atom %)

d. The liquid poison solution temperature shall not be less than the temperature presented in Figure 3.1.2.b.

e. If Specifications "a" through "d" are not met, initiate normal orderly shutdown within one hour.

Remove the squibs from the valves and verify that no deterioration has occurred by actual field firing of the removed squibs. In addition, field fire one squib from the batch of replacements.

Disassemble and inspect the squib-operated valves to verify that valve deterioration has not occurred.

(2) In accordance with the Surveillance Frequency Control Program -

Demineralized water shall be recycled to the test tank. Pump discharge pressure and minimum flow rate shall be verified.

b. Boron Solution Checks:

(1) In accordance with the Surveillance Frequency Control Program -

Boron concentration shall be determined.

(2) In accordance with the Surveillance Frequency Control Program -

Solution volume shall be checked. In addition, the sodium pentaborate concentration shall be determined and conformance with the requirements of the equivalency equation shall be checked any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.1.2.b.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(3) In accordance with the Surveillance Frequency Control Program - The solution temperature shall be checked. (4) In accordance with the Surveillance Frequency Control Program - Verify enrichment by analysis. c. Surveillance with Inoperable Components When a component becomes inoperable its redundant component shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to the operating status of the emergency cooling system.

Objective:

To assure the capability of the emergency cooling system to cool the reactor coolant in the event the normal reactor heat sink is not available.

Specification:

a. During power operating conditions and whenever the reactor coolant temperature is greater than 212°F except for hydrostatic testing with the reactor not critical, both emergency cooling systems shall be operable except as specified in 3.1.3.b.

b. If one emergency cooling system becomes inoperable, Specification 3.1.3.a shall be considered fulfilled, provided that the inoperable system is returned to an operable condition within 7 days and the additional surveillance required in 4.1.3.f is performed.

4.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to periodic testing requirements for the emergency cooling system.

Objective:

To assure the capability of the emergency cooling system for cooling of the reactor coolant.

Specification:

The emergency cooling system surveillance shall be performed as indicated below:

a. In accordance with the Surveillance Frequency Control Program -

The system heat removal capability shall be determined.

b. In accordance with the Surveillance Frequency Control Program -

The shell side water level and makeup tank water level shall be checked.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. Make up water shall be available from the two gravity feed makeup Water tanks.	c. In accordance with the Surveillance Frequency Control Program - The makeup tank level control valve shall be manually opened and closed.
d. During Power Operating Conditions, each emergency cooling system high point vent to torus shall be operable. 1. With a vent path for one emergency cooling system inoperable, restore the vent path to an operable condition within 30 days. 2. With vent paths for both emergency cooling systems inoperable, restore one vent path to an operable condition with 14 days and both vent paths within 30 days.	d. In accordance with the Surveillance Frequency Control Program - The area temperature shall be checked. e. In accordance with the Surveillance Frequency Control Program - Automatic actuation and functional system testing shall be performed in accordance with the Surveillance Frequency Control Program and whenever major repairs are completed on the system.
e. If Specification 3.1.3.a, b, c, or d are not met, a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the cold shutdown conditions within ten hours.	Each emergency cooling vent path shall be demonstrated operable by cycling each power-operated valve (05-01R, 05-11, 05-12, 05-04R, 05-05 and 05-07) in the vent path through one complete cycle of full travel and verifying that all manual valves are in the open position. f. Surveillance with an Inoperable System - When one of the emergency cooling systems is inoperable, the level control valve and the motor-operated isolation valve in the operable system shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.4 CORE SPRAY SYSTEM

Applicability:

Applies to the operating status of the core spray systems.

Objective:

To assure the capability of the core spray systems to cool reactor fuel in the event of a loss-of-coolant accident.

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, each of the two core spray systems shall be operable except as specified in Specifications b and c below.

b. If a redundant component of a core spray system becomes inoperable, that system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

c. If a redundant component in each of the core spray systems becomes inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

4.1.4 CORE SPRAY SYSTEM

Applicability:

Applies to the periodic testing requirements for the core spray systems.

Objective:

To verify the operability of the core spray systems.

Specification:

The core spray system surveillance shall be performed as indicated below.

a. In accordance with the Surveillance Frequency Control Program automatic actuation of each subsystem in each core spray system shall be demonstrated.

b. In accordance with the Surveillance Frequency Control Program pump operability shall be checked.

c. In accordance with the Surveillance Frequency Control Program the operability of power-operated valves required for proper system operation shall be checked.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
d. If Specifications a, b and c are not met, a normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.	d. (Deleted)
e. During reactor operation, except during core spray system surveillance testing, core spray isolation valves 40-02 and 40-12 shall be in the open position and the associated valve motor starter circuit breakers for these valves shall be locked in the off position. In addition, redundant valve position indication shall be available in the control room.	e. <u>Surveillance with Inoperable Components</u> When a component becomes inoperable its redundant component or system shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.
f. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, two core spray subsystems shall be operable except as specified in g and h below.	f. With a core spray subsystem suction from the CST, CST level shall be checked in accordance with the Surveillance Frequency Control Program.
g. If one of the above required subsystems becomes inoperable, restore at least two subsystems to an operable status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.	g. In accordance with the Surveillance Frequency Control Program when the reactor coolant temperature is greater than 212°F, verify that the piping system between valves 40-03, 13 and 40-01, 09, 10, 11 is filled with water.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.1.5 <u>SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)</u>	4.1.5 <u>SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)</u>
<u>Applicability</u>:	<u>Applicability</u>:
Applies to the operational status of the solenoid-actuated relief valves.	Applies to the periodic testing requirements for the solenoid-actuated pressure relief valves.
<u>Objective</u>:	<u>Objective</u>:
To assure the capability of the solenoid-actuated pressure relief valves to provide a means of depressurizing the reactor in the event of a small line break to allow full flow of the core spray system.	To assure the operability of the solenoid-actuated pressure relief valves to perform their intended functions.
<u>Specification</u>:	<u>Specification</u>:
	The solenoid-actuated pressure relief valve surveillance shall be performed as indicated below.
a. During power operating condition whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, all six solenoid-actuated pressure relief valves shall be operable.	a. In accordance with the Surveillance Frequency Control Program, verify each valve actuator strokes when manually actuated.
b. If specification 3.1.5a above is not met, the reactor coolant pressure and the reactor coolant temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.	b. In accordance with the Surveillance Frequency Control Program, automatic initiation shall be demonstrated.

LIMITING CONDITION FOR OPERATION

3.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

Applicability:

Applies to the operational status of the control rod drive pump coolant injection system.

Objective:

To assure the capability of the control rod drive pump coolant injection system to:

Provide core cooling in the event of a small line break, and

Provide coolant makeup in the event of reactor coolant leakage (see LCO 3.2.5).

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, the control rod drive pump coolant injection system shall be operable except as specified in "b" below.

SURVEILLANCE REQUIREMENT

4.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the control rod drive pump coolant injection system.

Objective:

To assure the capability of the control rod drive pump coolant injection system in performing its intended functions.

Specification:

The control rod drive pump coolant injection system surveillance shall be performed as indicated below.

a. In accordance with the Surveillance Frequency Control Program -

Automatic starting of each pump shall be demonstrated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. If a redundant component becomes inoperable, the control rod drive pump coolant injection system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.	b. <u>In accordance with the Surveillance Frequency Control Program -</u> Pump flow rate shall be determined.
c. If Specifications "a" or "b" above are not met, the reactor coolant temperature shall be reduced to 212°F or less within ten hours.	c. <u>Surveillance with Inoperable Components</u> When a component becomes inoperable, its redundant component shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

LIMITING CONDITION FOR OPERATION

3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is

SURVEILLANCE REQUIREMENT

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of axial location and average planar exposure shall be determined in accordance with the Surveillance Frequency Control Program during reactor operation at ≥25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION

not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.

b. <u>Linear Heat Generation Rate</u> (LHGR)

During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the limiting value specified in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded at any location, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR at all locations is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until LHGR at all locations is within the prescribed limits.

c. <u>Minimum Critical Power Ratio (MCPR)</u>

During power operation, the MCPR for all fuel at rated power and flow shall be within the limit provided in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within

SURVEILLANCE REQUIREMENT

b. <u>Linear Heat Generation Rate</u> (LHGR)

The LHGR as a function of core height shall be checked in accordance with the Surveillance Frequency Control Program during reactor operation at ≥25% rated thermal power.

c. <u>Minimum Critical Power Ratio</u> (MCPR)

(1) MCPR shall be determined in accordance with the Surveillance Frequency Control Program during reactor power operation at >25% rated thermal power.

(2) MCPR operating limit shall be determined within 72 hours of completing scram time testing as required in Specification 4.1.1(c).

LIMITING CONDITION FOR OPERATION

the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit. For core flows other than rated, the MCPR limit shall be the limit identified above times K_f where K_f is provided in the Core Operating Limits Report.

d. Power Flow Relationship During Operation

This power/flow relationship shall not exceed the limiting values shown in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until the power/flow relationship is within the prescribed limits.

e. Partial Loop Operation

During power operation, partial loop operation is permitted provided the following conditions are met.

SURVEILLANCE REQUIREMENT

d. Power Flow Relationship

Compliance with the power flow relationship in Section 3.1.7.d shall be determined in accordance with the Surveillance Frequency Control Program during reactor operation.

e. Partial Loop Operation

Under partial loop operation, surveillance requirements 4.1.7, a, b, c and d above are applicable.

LIMITING CONDITION FOR OPERATION

3.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the operational status of the high pressure coolant injection system.

Objective:

To assure the capability of the high pressure coolant injection system to cool reactor fuel in the event of a loss-of-coolant accident.

Specification:

a. During the power operating condition* whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, the high pressure coolant injection system shall be operable except as specified in Specification "b" below.

b. If a redundant component of the high pressure coolant injection system becomes inoperable, the high pressure coolant injection shall be considered operable provided that the component is returned to an operable condition within 15 days and the additional surveillance required is performed.

* One Feedwater Pump blocking valve in one HPCI pump train may be closed during reactor startup when core power is equal to or less than 25% of rated thermal power.

SURVEILLANCE REQUIREMENT

4.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the high pressure coolant injection system.

Objective:

To verify the operability of the high pressure coolant injection system.

Specification:

The high pressure coolant injection surveillance shall be performed as indicated below:

a. In accordance with the Surveillance Frequency Control Program -

Automatic start-up of the high pressure coolant injection system shall be demonstrated.

b. In accordance with the Surveillance Frequency Control Program -

Pump operability shall be determined.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specification "a" and "b" are not met, a normal orderly shutdown shall be initiated within one hour and reactor coolant pressure and temperature shall be reduced to less than 110 psig and saturation temperature within 24 hours.	c. <u>Surveillance with Inoperable Components</u> When a component becomes inoperable, its redundant component shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.

LIMITING CONDITION FOR OPERATION

3.2.3 COOLANT CHEMISTRY

Applicability:

Applies to the reactor coolant system chemical requirements.

Objective:

To assure the chemical purity of the reactor coolant water.

Specification:

a. The reactor coolant water shall not exceed the following limits for > 24 hours with the coolant temperature ≥200 degrees F and reactor thermal power ≤ 10%, or a shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.

Conductivity	1 μmho/cm*
Chloride ion	100 ppb
Sulfate ion	100 ppb

* During Noble Metal Chemical Addition (NMCA), the limit is 20 μmho/cm. Post NMCA, the conductivity limit is 2 μmho/cm for up to a 5 month period at power operation.

SURVEILLANCE REQUIREMENT

4.2.3 COOLANT CHEMISTRY

Applicability:

Applies to the periodic testing requirements of the reactor coolant chemistry.

Objective:

To determine the chemical purity of the reactor coolant water.

Specification:

Samples shall be taken and analyzed for conductivity, chloride and sulfate ion content in accordance with the Surveillance Frequency Control Program. In addition, if the conductivity becomes abnormal (other than short term spikes) as indicated by the continuous conductivity monitor, samples shall be taken and analyzed within 8 hours.

When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken and analyzed for conductivity, chloride and sulfate ion content at least once per 8 hours.

LIMITING CONDITION FOR OPERATION

3.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

Applicability:

Applies to the limits on reactor coolant specific activity.

Objective:

To assure that in the event of a reactor coolant system line break outside the drywell permissible doses are not exceeded.

Specification:

a. During the power operating and hot shutdown conditions, the specific activity of the reactor coolant shall be limited to Dose Equivalent I-131 specific activity $\leq$ 0.2 μCi/gm.

b. If reactor coolant specific activity is $>$ 0.2 μCi/gm and $\leq$ 4.0 μCi/gm Dose Equivalent I-131, determine the Dose Equivalent I-131 once per 4 hours and restore Dose Equivalent I-131 to within the limit of Specification 3.2.4.a within 48 hours.

SURVEILLANCE REQUIREMENT

4.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

Applicability:

Applies to the periodic testing requirements of the reactor coolant specific activity.

Objective:

To assure that limits on coolant specific activity are not exceeded.

Specification:

a. When the unit is in the power operating condition, verify that reactor coolant Dose Equivalent I-131 specific activity is $\leq$ 0.2 μCi/gm in accordance with the Surveillance Frequency Control Program.

b. Verify that reactor coolant Dose Equivalent I-131 specific activity is $\leq$ 0.2 μCi/gm within 24 hours prior to raising the reactor coolant temperature > 215°F, with the reactor not critical, and with primary containment integrity not established.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the limits on reactor coolant system leakage rate and leakage detection systems.

Objective:

To assure that the makeup capability provided by the control rod drive pump is not exceeded.

Specification:

a. Any time irradiated fuel is in the reactor vessel and the reactor temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:

1. Five gallons per minute unidentified leakage.

2. A two gallon per minute increase in unidentified leakage within any period of 24 hours or less.

3. Twenty-five gallons per minute total leakage (identified plus unidentified) averaged over any 24 hour period.

4.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the monitoring of reactor coolant system leakage.

Objective:

To determine the reactor coolant system leakage rate and assure that the leakage limits are not exceeded.

Specification:

a. A check of the reactor coolant leakage shall be made in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

b. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, at least one of the leakage measurement channels associated with each sump (one for the drywell floor drain and one for the equipment drain) shall be operable.

If the conditions a or b cannot be met, the reactor will be placed in the cold shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENT

b. The following surveillance shall be performed on each leakage detection system:

(1) An instrument calibration in accordance with the Surveillance Frequency Control Program.

(2) An instrument functional test in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

3.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines connected to the reactor coolant system.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system, and to minimize potential leakage paths from the primary containment in the event of a loss-of-coolant accident.

Specification:

a. Whenever fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, all reactor coolant system isolation valves on lines connected to the reactor coolant system shall be operable except as specified in Specification 3.2.7.b below.

b. In the event any isolation valve becomes inoperable whenever fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, the system shall be considered operable provided that within 4 hours at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition, except as noted in Specification 3.1.1.e.

SURVEILLANCE REQUIREMENT

4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the periodic testing requirement for the reactor coolant system isolation valves.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system, and to limit potential leakage paths from the primary containment in the event of a loss-of-coolant accident.

Specification:

The reactor coolant system isolation valves surveillance shall be performed as indicated below.

a. In accordance with the Surveillance Frequency Control Program the operable automatically initiated power-operated isolation valves shall be tested for automatic initiation and closure times.

b. Additional surveillances shall be performed as required by Specification 6.5.4.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. If Specifications 3.2.7a and b above are not met, initiate normal orderly shutdown within one hour and have reactor in the cold shutdown condition within ten hours.

d. Whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, the isolation valves on the shutdown cooling system lines connected to the reactor coolant system shall be operable except as specified in Specification 3.2.7.e below.

e. In the event any shutdown cooling system isolation valve becomes inoperable whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, the system shall be considered operable provided that, within 4 hours, at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

f. If Specifications 3.2.7.d and 3.2.7.e above are not met, either:

(1) Immediately initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs); or

(2) Immediately initiate action to restore the valve(s) to operable status.

c. In accordance with the Surveillance Frequency Control Program the feedwater and main-steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

d. At least once per plant cold shutdown the feedwater and main steam line power-operated isolation valves shall be fully closed and reopened, unless this test has been performed within the previous 92 days.

LIMITING CONDITION FOR OPERATION

3.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the operational status of the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure below the safety limit in the event of rapid reactor isolation and failure of all pressure relieving devices.

Specification:

a. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than saturation temperature all nine of the safety valves shall be operable.

b. If specification 3.2.8a is not met, the reactor coolant pressure and temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.

SURVEILLANCE REQUIREMENT

4.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the periodic testing requirements for the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure to below the safety limit.

Specification:

In accordance with the Surveillance Frequency Control Program, the number of safety valves as determined by the IST Program Plan shall be removed, tested for set point and partial lift, and then returned to operation or replaced.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	b. In accordance with the Surveillance Frequency Control Program, verify each valve actuator strokes when manually actuated. c. In accordance with the Surveillance Frequency Control Program, relief valve setpoints shall be verified.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.1 OXYGEN CONCENTRATION	4.3.1 OXYGEN CONCENTRATION
Applicability:	Applicability:
Applies to the limit on oxygen concentration within the primary containment system.	Applies to the periodic testing requirement for the primary containment system oxygen concentration.
Objective:	Objective:
To assure that in the event of a loss-of-coolant accident any hydrogen generation will not result in a combustible mixture within the primary containment system.	To assure that the oxygen concentration within the primary containment system is within required limits.
Specification:	Specification:
a. The primary containment atmosphere shall be reduced to less than four percent by volume oxygen concentration with nitrogen gas whenever the reactor coolant pressure is greater than 110 psig and the reactor is in the power operating condition, except as specified in "b" and "c" below.	In accordance with the Surveillance Frequency Control Program, oxygen concentration shall be determined.

LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the interrelated parameters of pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the peak suppression chamber pressure does not exceed design values in the event of a loss-of-coolant accident.

Specification:

a. The downcomers in the suppression chamber shall have a minimum submergence of three and one half feet and a maximum submergence of four and one quarter feet whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required.

b. During normal power operation, suppression chamber water temperature shall be less than or equal to 85°F.

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the periodic testing of the pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the pressure suppression system pressure and suppression chamber water temperature and level are within required limits.

Specification:

a. In accordance with the Surveillance Frequency Control Program, the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

b. A visual inspection of the suppression chamber interior, including water line regions, shall be made in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines open to the free space of the primary containment.

Objective:

To assure that potential leakage paths from the primary containment in the event of a loss-of-coolant accident are minimized.

Specification:

a. Whenever the reactor coolant system temperature is greater than 215°F and primary containment integrity is required, all containment isolation valves on lines open to the free space of the primary containment shall be operable except as specified in 3.3.4b below.

b. In the event any isolation valve becomes inoperable the system shall be considered operable provided that within 4 hours at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

4.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the primary containment isolation valve system.

Objective:

To assure the operability of the primary containment isolation valves to limit potential leakage paths from the containment in the event of a loss-of-coolant accident.

Specification:

The primary containment isolation valves surveillance shall be performed as indicated below.

a. In accordance with the Surveillance Frequency Control Program, the operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure times.

b. In accordance with the Surveillance Frequency Control Program, all normally open power operated isolation valves shall be fully closed and reopened.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specifications 3.3.4 a and b are not met, the reactor coolant system temperature shall be reduced to a value less than 215°F within ten hours.	c. In accordance with the Surveillance Frequency Control Program, each instrument-line flow check valve will be tested for operability.

LIMITING CONDITION FOR OPERATION

3.3.6 VACUUM RELIEF

Applicability:

Applies to the operational status of the primary containment vacuum relief system.

Objective:

To assure the capability of the vacuum relief system in the event of a loss-of-coolant accident to:

a. Equalize pressures between the drywell and suppression chamber, and

b. Maintain containment pressure above the vacuum design values of the drywell and suppression chamber.

Specification:

a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated above. Suppression chamber - drywell vacuum breakers shall be considered operable if:

   (1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

SURVEILLANCE REQUIREMENT

4.3.6 VACUUM RELIEF

Applicability:

Applies to the periodic testing of the vacuum relief system.

Objective:

To assure the operability of the containment vacuum relief system to perform its intended functions.

Specification:

a. Periodic Operability Tests

   In accordance with the Surveillance Frequency Control Program and following any release of energy to the suppression chamber, each suppression chamber - drywell vacuum breaker shall be exercised. Operability of valves, position switches, and position indicators and alarms shall be verified monthly and following any maintenance on the valves and associated equipment.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(2) The valve can be closed by gravity, when released after being opened by remote or manual means, to within not greater than the equivalent of 0.06 inch at the bottom of the disk. (3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 0.06 inch at the bottom of the disk. b. Any drywell-suppression chamber vacuum breaker may be non-fully closed as indicated by the position indication and alarm systems provided that drywell to suppression chamber differential pressure decay rate is demonstrated to be not greater than 25% of the differential pressure decay rate for all vacuum breakers open the equivalent of 0.06 inch at the bottom of the disk.	b. <u>In accordance with the Surveillance Frequency Control Program</u> (1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open. (2) All suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (3) In accordance with the Surveillance Frequency Control Program, each vacuum breaker valve shall be visually inspected to ensure proper maintenance and operation. (4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. When it is determined that one or more vacuum breaker valves are not fully closed as indicated by the position indication system at a time when such closure is required, the apparently malfunctioning vacuum breaker valve shall be exercised and pressure tested as specified in 3.3.6 b immediately and every 15 days thereafter until appropriate repairs have been completed.

d. One drywell-suppression chamber vacuum breaker may be secured in the closed position.

e. If Specifications 3.3.6 a, b, c, or d cannot be met, the situation shall be corrected within 24 hours or the reactor shall be placed in a cold shutdown condition within 24 hours.

f. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u>

(1) The three pressure suppression chamber reactor building vacuum breaker systems shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air-operated vacuum breakers shall be $\leq$0.5 psid.

c. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u>

(1) The pressure suppression chamber-reactor building vacuum breaker systems and associated instrumentation, including set point, shall be checked for proper operation in accordance with the Surveillance Frequency Control Program.

(2) In accordance with the Surveillance Frequency Control Program, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.3.6.f(1) and each vacuum breaker shall be inspected and verified to meet design requirement.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.7 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the operating status of the containment spray system.

Objective:

To assure the capability of the containment spray system to limit containment pressure and temperature in the event of a loss-of-coolant accident.

Specification:

a. During all reactor operating conditions whenever reactor coolant temperature is greater than 215°F and fuel is in the reactor vessel and primary containment integrity is required; each of the two containment spray systems and the associated raw water cooling systems shall be operable except as specified in 3.3.7.b.

b. If a redundant component of a containment spray system becomes inoperable, Specification 3.3.7.a shall be considered fulfilled, provided that the component is returned to an operable condition within 15 days and that the additional surveillance required is performed.

4.3.7 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the testing of the containment spray system.

Objective:

To verify the operability of the containment spray system.

Specification:

The containment spray system surveillance shall be performed as indicated below:

a. Containment Spray Pumps

(1) In accordance with the Surveillance Frequency Control Program, automatic startup of the containment spray pump shall be demonstrated.

(2) In accordance with the Surveillance Frequency Control Program, pump operability shall be checked.

b. Nozzles

Following maintenance that could result in nozzle blockage, a test shall be performed on the spray nozzles.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If a redundant component in each of the containment spray systems or their associated raw water systems become inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and that the additional surveillance required is performed.	c. Raw Water Cooling Pumps In accordance with the Surveillance Frequency Control Program, manual startup and operability of the raw water cooling pumps shall be demonstrated.
d. If a containment spray system or its associated raw water system becomes inoperable and all the components are operable in the other systems, the reactor may remain in operation for a period not to exceed 7 days.	d. Surveillance with Inoperable Components When a component or system becomes inoperable its redundant component or system shall be verified to be operable immediately and in accordance with the Surveillance Frequency Control Program thereafter.
e. If Specifications "a" or "b" are not met, shutdown shall begin within one hour and the reactor coolant shall be below 215°F within ten hours. If both containment spray systems become inoperable the reactor shall be in the cold shutdown condition within ten hours and no work shall be performed on the reactor which could result in lowering the reactor water level to more than six feet, three inches (-10 inches indicator scale) below minimum normal water level (Elevation 302'9").	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
f. The containment spray system shall be considered operable by verifying that lake water temperature does not exceed 83°F. g. If specification "f" cannot be met commence shutdown within one hour and be in hot shutdown within 8 hours and cold shutdown within 24 hours.	f. Lake Water Temperature Record in accordance with the Surveillance Frequency Control Program and at least once per 8 hours when latest recorded water temperature is greater than or equal to 75°F and at least once per 4 hours when the latest recorded water temperature is greater than or equal to 79°F.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.4.1 LEAKAGE RATE

Applicability:

Applies to the leakage rate of the secondary containment.

Objective:

To specify the requirements necessary to limit exfiltration of fission products released to the secondary containment as a result of an accident.

Specification:

At all times when secondary containment integrity is required, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 1600 cfm. If this cannot be met after a routine surveillance check, then the actions listed below shall be taken:

a. Suspend any of the following activities:

1. Handling of recently irradiated fuel in the reactor building.
2. Irradiated fuel cask operations in the reactor building.
3. Operations with a potential for draining the reactor vessel (OPDRVs).

b. Restore the reactor building leakage rates to within specified limits within 4 hours or initiate normal orderly shutdown and be in a cold shutdown condition within 10 hours.

4.4.1 LEAKAGE RATE

Applicability:

Applies to the periodic testing requirements of the secondary containment leakage rate.

Objective:

To assure the capability of the secondary containment to maintain leakage within allowable limits.

Specification:

In accordance with the Surveillance Frequency Control Program - isolate the reactor building and start emergency ventilation system fan to demonstrate negative pressure in the building relative to external static pressure. The fan flow rate shall be varied so that the building internal differential pressure is at least as negative as that on Figure 3.4.1 for the wind speed at which the test is conducted. The fan flow rate represents the reactor building leakage referenced to zero mph with building internal pressure at least 0.25 inch of water less than atmospheric pressure. The test shall be done at wind speeds less than 20 miles per hour.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

Applicability:

Applies to the operational status of the reactor building isolation valves.

Objective:

To assure that fission products released to the secondary containment are discharged to the environment in a controlled manner using the emergency ventilation system.

Specification:

a. The normal Ventilation System isolation valves shall be operable at all times when secondary containment integrity is required.

b. If Specification 3.4.2.a is not met, then the actions listed below shall be taken:

1. The reactor shall be in the cold shutdown condition within ten hours.

2. Suspend any of the following activities:

    a. Handling of recently irradiated fuel in the reactor building,

    b. Irradiated fuel cask handling operations in the reactor building,

    c. Operations with a potential for draining the reactor vessel (OPDRVs).

4.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the reactor building isolation valves.

Objective:

To assure the operability of the reactor building isolation valves.

Specification:

In accordance with the Surveillance Frequency Control Program, automatic initiation of valves shall be checked.

LIMITING CONDITION FOR OPERATION

3.4.3 ACCESS CONTROL

Applicability:

Applies to the access control to the reactor building.

Objective:

To specify the requirements necessary to assure the integrity of the secondary containment system.

Specification:

a. At all times when secondary containment integrity is required, the following conditions will be met:

    1. Only one door in each of the double-doored access ways shall be opened at one time.

    2. Only one door or closeup of the railroad bay shall be opened at one time.

    3. The core spray and containment spray pump compartments' doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

SURVEILLANCE REQUIREMENT

4.4.3 ACCESS CONTROL

Applicability:

Applies to the periodic checking of the condition of portions of the reactor building.

Objective:

To assure that pump compartments are properly closed at all times and to assure the integrity of the secondary containment system by verifying that reactor building access doors are closed, as required by Specifications 3.4.3.a.1 and 3.4.3.a.2.

Specification:

a. The core and containment spray pump compartments shall be checked in accordance with the Surveillance Frequency Control Program and after each entry.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. If these conditions cannot be met, then the actions listed below shall be taken: 1. If in the power operating condition, restore reactor building integrity within 4 hours or be in at least the hot shutdown condition within the next 12 hours and in the cold shutdown condition within the following 24 hours. OR If the reactor coolant system temperature is above 215°F, restore reactor building integrity within 4 hours or be in cold shutdown within the following 24 hours. 2. Suspend any of the following activities: a. Handling of recently irradiated fuel in the reactor building, b. Irradiated fuel cask handling operations in the reactor building, c. Operations with a potential for draining the reactor vessel (OPDRVs).	b. Verify in accordance with the Surveillance Frequency Control Program that: 1. At least one door in each access to the secondary containment is closed. 2. At least one door or closeup of the railroad bay is closed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.4.4 EMERGENCY VENTILATION SYSTEM

Applicability:

Applies to the operating status of the emergency ventilation system.

Objective:

To assure the capability of the emergency ventilation system to minimize the release of radioactivity to the environment in the event of an incident within the primary containment or reactor building.

Specification:

a. Except as specified in Specification 3.4.4e below, both circuits of the emergency ventilation system shall be operable at all times when secondary containment integrity is required.

b. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

4.4.4 EMERGENCY VENTILATION SYSTEM

Applicability:

Applies to the testing of the emergency ventilation system.

Objective:

To assure the operability of the emergency ventilation system.

Specification:

Emergency ventilation system surveillance shall be performed as indicated below:

a. In accordance with the Surveillance Frequency Control Program, the following conditions shall be demonstrated:

(1) Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system rated flow rate ($\pm$ 10%).

(2) Deleted

LIMITING CONDITION FOR OPERATION

c. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 at 30°C and 95% R.H.

d. Fans shall be shown to operate within ±10% design flow.

e. During reactor operation, including when the reactor coolant system temperature is above 215°F, from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other emergency ventilation circuit shall be operable.

During handling of recently irradiated fuel in the reactor building, handling of an irradiated fuel cask in the reactor building, and operations with a potential for draining the reactor vessel (OPDRVs), from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, recently irradiated fuel handling in the reactor building, irradiated fuel cask handling in the reactor building, or OPDRVs are permissible during the succeeding seven days unless such circuit is sooner made operable, provided that

SURVEILLANCE REQUIREMENT

b. The tests and sample analysis of Specification 3.4.4b, c and d shall be performed in accordance with the Surveillance Frequency Control Program, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release in any ventilation zone communicating with the system.

c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

e. Each circuit shall be operated at least 15 minutes in accordance with the Surveillance Frequency Control Program.

f. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at and in conformance with each test performed for compliance with Specification 4.4.4b and Specification 3.4.4b.

LIMITING CONDITION FOR OPERATION

during such seven days all active components of the other emergency ventilation circuit shall be operable. Recently irradiated fuel handling in the reactor building, irradiated fuel cask handling in the reactor building, or OPDRVs may continue beyond seven days provided the operable emergency ventilation circuit is in operation.

f. If these conditions cannot be met, within 36 hours, the reactor shall be placed in a condition for which the emergency ventilation system is not required.

SURVEILLANCE REQUIREMENT

g. In accordance with the Surveillance Frequency Control Program, automatic initiation of each branch of the emergency ventilation system shall be demonstrated.

h. In accordance with the Surveillance Frequency Control Program, manual operability of the bypass valve for filter cooling shall be demonstrated.

i. When one circuit of the emergency ventilation system becomes inoperable all active components in the other emergency ventilation circuit shall be verified to be operable within two hours and in accordance with the Surveillance Frequency Control Program thereafter.

LIMITING CONDITION FOR OPERATION

3.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the operating status of the control room air treatment system and Control Room Envelope (CRE) boundary.

---------------------------NOTE---------------------------
The CRE boundary may be opened intermittently under administrative control.

Objective:

To assure the capability of the control room air treatment system to minimize the amount of radioactivity or other gases entering the control room in the event of an incident.

Specification:

a. Except as specified below, the control room air treatment system shall be operable for the following conditions:

   1. Power operating condition, and whenever the reactor coolant system temperature is greater than 212°F,

   2. Whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, and

   3. During operations with a potential for draining the reactor vessel (OPDRVs).

b. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

SURVEILLANCE REQUIREMENT

4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the testing of the control room air treatment system and CRE boundary.

Objective:

To assure the operability of the control room air treatment system.

Specification:

a. In accordance with the Surveillance Frequency Control Program, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 1.5 inches of water at system design flow rate (± 10%).

b. The tests and sample analysis of Specification 3.4.5b, c and d shall be performed in accordance with the Surveillance Frequency Control Program, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release i any ventilation zone communicating with the system.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION

c. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodine removal when tested in accordance with ASTM D3803-1989 at 30°C and 95% R.H.

d. Fans shall be shown to operate within ±10% design flow.

e. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, except for an inoperable CRE boundary during the power operating condition, restore the system to operable within the succeeding seven days.

f. If the control room air treatment system is made or found to be inoperable due to an inoperable CRE boundary during the power operating condition: immediately initiate action to implement mitigating actions; within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits; and within 90 days, restore the CRE boundary to operable status.

g. If Specifications 3.4.5.e or 3.4.5.f cannot be met during the power operating condition, or when reactor coolant system temperature is greater than 212°F, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours.

h. If Specification 3.4.5.e cannot be met whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, or during OPDRVs: immediately suspend handling of recently irradiated fuel or the irradiated fuel cask in the reactor building; and immediately initiate action to suspend OPDRVs.

SURVEILLANCE REQUIREMENT

c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

e. The system shall be operated at least 15 minutes in accordance with the Surveillance Frequency Control Program.

f. In accordance with the Surveillance Frequency Control Program, automatic initiation of the control room air treatment system shall be demonstrated.

g. In accordance with the frequency and specifications of the Control Room Envelope Habitability Program, perform required CRE unfiltered air inleakage testing.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5.1 SOURCE RANGE MONITORS

Applicability:

Applies to the operating status of the source range monitors.

Objective:

To assure the capability of the source range monitors to provide neutron flux indication required for reactor shutdown and startup and refueling operations.

Specification:

Whenever the reactor is in the shutdown, refueling or power operating conditions (unless the IRM's or APRM's are on scale) or whenever core alterations are being made at least three SRM channels will be operable except as noted in Specification 3.5.3. To be considered operable, the following conditions must be satisfied:

a. Inserted to normal operating level and available for monitoring the core. May be withdrawn as long as a minimum count rate of 100 cps is maintained.

4.5.1 SOURCE RANGE MONITORS

Applicability:

Applies to the periodic testing of the source range monitors.

Objective:

To assure the operability of the source range monitors to monitor low-level neutron flux.

Specification:

The source range monitoring system surveillance will be performed as indicated below.

In accordance with the Surveillance Frequency Control Program - check in-core to out-of-core signal ratio and minimum count rate.

LIMITING CONDITION FOR OPERATION

3.5.2 REFUELING PLATFORM INTERLOCK

Applicability:

Applies to the refueling platform on interlocks.

Objective:

To assure that a loaded refueling platform hoist is never over the core when one or more control rods are withdrawn.

Specification:

During the refueling condition with the mode switch in the "refuel" position the following interlocks must be operative:

a. Control rod withdrawal block with a fuel assembly on the hoist over the reactor core.

b. With a control rod withdrawn from the core the refuel platform, if loaded with a fuel assembly, is blocked from travelling over the core.

c. If the interlocks for either "a" or "b" or both are not operable, double procedural control will be used to ensure that "a" and "b" are met.

SURVEILLANCE REQUIREMENT

4.5.2 REFUELING PLATFORM INTERLOCK

Applicability:

Applies to the periodic testing requirements for the refueling platform interlocks.

Objective:

To assure the operability of the refueling platform interlock.

Specification:

The refueling platform interlocks shall be tested prior to any fuel handling with the head off the reactor vessel, and in accordance with the Surveillance Frequency Control Program thereafter until no longer required and following any repair work associated with the interlocks.

LIMITING CONDITION FOR OPERATION

3.6.1 MECHANICAL VACUUM PUMP ISOLATION

a. (Deleted)

b. The mechanical vacuum pump line shall be capable of automatic isolation by closure of the air-operated valve upstream of the pumps. The signal to initiate isolation shall be from high radioactivity (five times normal) in the main steam line.

SURVEILLANCE REQUIREMENT

4.6.1 MECHANICAL VACUUM PUMP ISOLATION

a. (Deleted)

b. In accordance with the Surveillance Frequency Control Program (prior to startup), verify automatic securing and isolation of the mechanical vacuum pump.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs a safety function.

Objective:

To assure the operability of the instrumentation required for safe operation.

Specification:

a. The set points, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.6.2a to 3.6.2l.

If the requirements of a table are not met, the actions listed below for the respective type of instrumentation shall be taken.

(1) Instrumentation that initiates scram - control rods shall be inserted, unless there is no fuel in the reactor vessel.

4.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs a safety function.

Objective:

To verify the operability of protective instrumentation.

Specification:

a. Sensors and instrument channels shall be checked, tested and calibrated at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Tables 4.6.2a to 4.6.2l.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. During operation with the Core Maximum Fraction of Limiting Power Density (CMFLPD) greater than the Fraction of Rated Thermal Power (FRTP), either: (1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2a; or (2) The APRM gain shall be adjusted in accordance with Specification 2.1.2a; or (3) The power distribution shall be changed such that the CMFLPD no longer exceeds FRTP.	c. During reactor power operation at ≥25 percent rated thermal power, the Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked in accordance with the Surveillance Frequency Control Program and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified by Specification 2.1.2a.

TABLE 4.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Manual Scram	None	Note 1	None
(2)	High Reactor Pressure	None	Note 1(l)	Note 1(l)
(3)	High Drywell Pressure	None	Note 1(l)	Note 1(l)
(4)	Low Reactor Water Level	Note 1	Note 1(l)	Note 1(l)
(5)	High Water Level Scram Discharge Volume	None	Note 1	Note 1
(6)	Main-Steam-Line Isolation Valve Position	None	Note 1	Note 1
(7)	Deleted			

TABLE 4.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(8) Shutdown Position of Reactor Mode Switch	None	Once during each major refueling outage	None
(9) Neutron Flux			
(a) IRM			
(i) Upscale	Note 1[f]	Note 1[g]	Note 1[n]
(ii) Inoperative	Note 1[f]	Note 1[g]	Note 1[n]
(b) APRM			
(i) Upscale	None	Note 1	Note 1[m] Note 1[n]
(ii) Inoperative	None	Note 1	None
(10) Turbine Stop Valve Closure	None	Note 1	Note 1
(11) Generator Load Rejection	None	Note 1	Note1

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed in the refuel and shutdown positions of the reactor mode switch with a keylock switch.

(c) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi, or for the purpose of performing reactor coolant system pressure testing and/or control rod scram time testing with the reactor mode switch in the refuel position.

(d) No more than one of the four IRM inputs to each trip system shall be bypassed.

(e) No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. A Traversing In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing.

(f) Verify SRM/IRM channels overlap during startup after the mode switch has been placed in startup. Verify IRM/APRM channels overlap at least 1/2 decade during entry into startup from run (normal shutdown) if not performed within the previous 7 days.

(g) Within 24 hours before startup, if not performed within the previous 7 days. Not required to be performed during shutdown until 12 hours after entering startup from run.

(h) Each of the four isolation valves has two limit switches. Each limit switch provides input to one of two instrument channels in a single trip system.

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

(k) May be bypassed when placing the reactor mode switch in the SHUTDOWN position and all control rods are fully inserted.

(l) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(m) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during reactor operation when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the difference is greater than +2.0/-1.9% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 2.1.2a shall not be included in determining the difference.

(n) Neutron detectors are excluded.

(n) Deleted.

(o) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same trip system is monitoring that parameter.

With one channel required by Table 3.6.2a inoperable in one or more Parameters, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours.

With two or more channels required by Table 3.6.2a inoperable in one or more Parameters:

1. Within one hour, verify sufficient channels remain Operable or tripped* to maintain trip capability for the Parameter, and

2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

3. Within 12 hours, restore the inoperable channels in the other trip system to an Operable status or tripped*.

Otherwise, take the ACTION required by Specification 3.6.2a for that Parameter.

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to Operable status within the required time, the ACTION required by Specification 3.6.2a for the parameter shall be taken.

** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

(p) May be bypassed during reactor coolant system pressure testing and/or control rod scram time testing.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2a

TABLE 4.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
PRIMARY COOLANT ISOLATION (Main Steam, Cleanup and Shutdown Cooling)			
(1) Low-Low Reactor Water Level	Note 1	Note 1[(d)]	Note 1[(d)]
(2) Manual	---	Note 1	---
MAIN-STEAM-LINE ISOLATION			
(3) High Steam Flow Main-Steam Line	Note 1	Note 1[(d)]	Note 1[(d)]
(4) Deleted			
(5) Low Reactor Pressure	Note 1	Note 1[(d)]	Note 1[(d)]

TABLE 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(6)	Low-Low-Low Condenser Vacuum	None	Note 1	Note 1
(7)	High Temperature Main-Steam-Line Tunnel	None	Note 1	Note 1
CLEANUP SYSTEM ISOLATION				
(8)	High Area Temperature	Note 1	Note 1	Note 1
SHUTDOWN COOLING SYSTEM ISOLATION				
(9)	High Area Temperature	Note 1	Note 1	Note 1

TABLE 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter		Sensor Check	Instrument Channel Test	Instrument Channel Calibration
CONTAINMENT ISOLATION				
(10)	Low-Low Reactor Water Level	Note 1	Note 1[(d)]	Note 1[(d)]
(11)	High Drywell Pressure	Note 1	Note 1[(d)]	Note 1[(d)]
(12)	Manual	---	Note 1	---

NOTES FOR TABLES 3.6.2b and 4.6.2b

(a) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(b) May be bypassed when necessary for containment inerting.

(c) May be bypassed in the shutdown mode whenever the reactor coolant system temperature is less than 215°F.

(d) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(e) Deleted.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable Channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within

   a. 12 hours for Parameters common to SCRAM Instrumentation, and
   b. 24 hours for Parameters not common to SCRAM Instrumentation.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable Channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within

      (1) 12 hours for Parameters common to SCRAM Instrumentation, and
      (2) 24 hours for Parameters not common to SCRAM Instrumentation.

   or

   b. take the ACTION required by Specification 3.6.2a for that Parameter.

(g) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

(h) Only applicable during startup mode while operating in IRM range 10.

(i) May be bypassed in the cold shutdown condition.

(j) In the cold shutdown and refueling conditions, only one Operable Trip System is required provided shutdown cooling system integrity is maintained. With one of the two required Operable Channels in the required Trip System not operable, place the inoperable channel in the tripped condition within 12 hours. Otherwise, either:

1. Immediately initiate action to restore the channel to operable status.

or

2. Immediately initiate action to isolate the shutdown cooling system.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2b.

TABLE 4.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
EMERGENCY COOLING INITIATION			
(1) High Reactor Pressure	None	Note 1[(c)]	Note 1[(c)]
(2) Low-Low Reactor Water Level	Note 1	Note 1[(c)]	Note 1[(c)]
EMERGENCY COOLING ISOLATION (for each of two systems)			
(3) High Steam Flow Emergency Cooling System	None	Note 1[(c)]	Note 1[(c)]

NOTES FOR TABLES 3.6.2c AND 4.6.2c

(a) Each of two differential pressure switches provide inputs to one instrument channel in each trip system.

(b) May be bypassed in the cold shutdown condition.

(c) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

(e) With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the action required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

(f) With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.
or
2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour
and
2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 24 hours.
   or
   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2c.

TABLE 4.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
START CORE SPRAY PUMPS			
(1) High Drywell Pressure	Note 1	Note 1[(c)]	Note 1[(c)]
(2) Low-Low Reactor Water Level	Note 1	Note 1[(c)]	Note 1[(c)]
OPEN CORE SPRAY DISCHARGE VALVES			
(3) Reactor Pressure and either (1) or (2) above	None	Note 1[(c)]	Note 1[(c)]

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed when necessary for performing major maintenance as specified in Specification 2.1.1.e.

(c) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(d) May be bypassed when necessary for integrated leak rate testing.

(e) The instrumentation that initiates the Core Spray System is not required to be operable, if there is no fuel in the reactor vessel.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2d.

TABLE 4.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Surveillance Requirement

		Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	a.	High Drywell Pressure	Note 1	Note 1[(b)]	Note 1[(b)]
	b.	Low-Low Reactor Water Level	Note 1	Note 1[(b)]	Note 1[(b)]

NOTES FOR TABLES 3.6.2e AND 4.6.2e

(a) May be bypassed in the shutdown mode whenever the reactor coolant temperature is less than 215°F.

(b) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip system in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2e.

TABLE 4.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
INITIATION			
(1) a. Low-Low-Low Reactor Water	None	Note 1[(c)]	Note 1[(c)]
and			
b. High Drywell Pressure	Note 1	Note 1[(c)]	Note 1[(c)]

NOTES FOR TABLES 3.6.2f AND 4.6.2f

(a) Both instrument channels in either trip system are required to be energized to initiate auto depressurization. One trip system is powered from power board 102 and the other trip system from power board 103.

(b) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(c) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2f.

TABLE 4.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Deleted			
(2)	IRM			
	a. Detector not in Startup Position	N/A	Note 1[g]	N/A
	b. Inoperative	N/A	Note 1[g]	N/A
	c. Downscale	N/A	Note 1[g]	Note 1[j]
	d. Upscale	N/A	Note 1[g]	Note 1[j]
(3)	APRM			
	a. Inoperative	None	Note 1	None
	b. Upscale (Biased by Recirculation Flow)	None	Note 1	Note 1[j]
	c. Downscale	None	Note 1	Note 1[j]
(4)	Deleted			

TABLE 4.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(5)	Refuel Platform and Hoists	---	(see 4.5.2)	---
(6)	Mode Switch in Shutdown	---	Note 1	---
(7)	Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	---	Note 1	---
(8)	Deleted			

(a) Deleted

(b) No more than one of the four IRM inputs to each instrument channel shall be bypassed. These signals may be bypassed when the APRMs are onscale.

(c) No more than one of the four APRM inputs to each instrument channel shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to only one APRM shall be bypassed in order for the APRM to be considered operable. In the Run mode of operation, bypass of two chambers from one radial core location in any one APRM shall cause that APRM to be considered inoperative. A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing. If one APRM in a quadrant is bypassed and meets all requirements for operability with the exception of the requirement of at least one operable chamber at each radial location, it may be returned to service and the other APRM in that quadrant may be removed from service for test and/or calibration only if no control rod is withdrawn during the calibration and/or test.

(d) Deleted

(e) Deleted

(f) One sensor provides input to each of two instrument channels. Each instrument channel is in a separate trip system.

(g) Within 24 hours before startup, if not performed within the previous 7 days. Not required to be performed during shutdown until 12 hours after entering startup from run.

(h) The actuation of either or both trip systems will result in a rod block.

(i) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Parameter.

(j) Neutron detectors are excluded.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2g.

TABLE 4.6.2h

VACUUM PUMP ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
MECHANICAL VACUUM PUMP			
High Radiation Main Steam Line	Note 1	Note 1	Note 1

NOTES FOR TABLES 3.6.2h and 4.6.2h

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 12 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 12 hours.
   or
   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2h.

TABLE 4.6.2i

DIESEL GENERATOR INITIATION

Surveillance Requirements

Parameter	Sensor Check	Instrument[a] Channel Test	Instrument[b] Channel Calibration
Loss of Power			
a. 4.16kV PB 102/103 Emergency Bus Undervoltage (Loss of Voltage)	NA	Note 1	Note 1
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	NA	Note 1	Note 1

(a) The instrument channel test demonstrate the operability of the instrument channel by simulating an undervoltage condition to verify that the tripping logic functions properly.

(b) The instrument channel calibration will demonstrate the operability of the instrument channel by simulating an undervoltage condition to verify that the tripping logic functions properly. In addition, a sensor calibration will be performed to verify the set points listed in Table 3.6.2.i.

(c) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2i.

TABLE 4.6.2j

EMERGENCY VENTILATION INITIATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	High Radiation Reactor Building Ventilation Duct	Note 1	Note 1	Note 1
(2)	High Radiation Refueling Platform	Note 1	(c)	Note 1

(a) This function shall be operable whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, and during operations with a potential for draining the reactor vessel (OPDRVs).

(d) Deleted.

(e) Immediately prior to when function is required and in accordance with the Surveillance Frequency Control Program thereafter until function is no longer required.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1) Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2) Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2j.

TABLE 4.6.2k

HIGH PRESSURE COOLANT INJECTION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low Reactor Water Level	Note 1	Note 1[(b)]	Note 1[(b)]
(2)	Automatic Turbine Trip	None	Note 1	None

NOTES FOR TABLES 3.6.2k AND 4.6.2k

(a) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(b) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2k.

TABLE 4.6.2I

CONTROL ROOM AIR TREATMENT SYSTEM INITIATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low-Low Reactor Water Level	Note 1	Note 1[(b)]	Note 1[(b)]
(2)	High Steam Flow Main-Steam Line	Note 1	Note 1[(b)]	Note 1[(b)]
(3)	High Temperature Main-Steam Line Tunnel	---	Note 1	Note 1
(4)	High Drywell Pressure	Note 1	Note 1[(b)]	Note 1[(b)]

NOTES FOR TABLES 3.6.2I AND 4.6.2I

(a) May be bypassed when necessary for containment inerting.

(b) The trip circuit will be calibrated and tested in accordance with the Surveillance Frequency Control Program, the primary sensor will be calibrated and tested in accordance with the Surveillance Frequency Control Program.

(c) May be bypassed in the cold shutdown condition.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2I.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the operational status of the emergency power sources.

Objective:

To assure the capability of the emergency power sources to provide the power required for emergency equipment in the event of a loss-of-coolant accident.

Specification:

a. For all reactor operating conditions except cold shutdown, there shall normally be available two 115 kv external lines, two diesel generator power systems and two battery systems, except as further specified in "b," "c," "d," "e," and "h" below.

b. One 115 kv external line may be de-energized provided two diesel-generator power systems are operable. If a 115 kv external line is de-energized, that line shall be returned to service within 7 days.

4.6.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the periodic testing requirements for the emergency power sources.

Objective:

To assure the operability of the emergency power sources to provide emergency power required in the event of a loss-of-coolant accident.

Specification:

The emergency power systems surveillance will be performed as indicated below. In addition, components on which maintenance has been performed will be tested.

a. In accordance with the Surveillance Frequency Control Program - test for automatic startup and pickup of load required for a loss-of-coolant accident.

b. In accordance with the Surveillance Frequency Control Program - manual start and operation at rated load shall be performed for a minimum time of one hour. Determine the specific gravity of each cell. Determine the battery voltage.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. One diesel-generator power system may be inoperable provided two 115 kv external lines are energized. If a diesel-generator power system becomes inoperable, it shall be returned to an operable condition within 14 days. In addition, if a diesel-generator power system becomes inoperable coincident with a 115 kv line de-energized, that diesel-generator power system shall be returned to an operable condition within 24 hours.	c. <u>In accordance with the Surveillance Frequency Control Program</u> – determine the cell voltage and specific gravity of the pilot cells of each battery.
d. If a reserve power transformer becomes inoperable, it shall be returned to service within seven days.	d. <u>Surveillance for startup with an inoperable diesel-generator</u> – prior to startup the operable diesel-generator shall be tested for automatic startup and pickup of the load required for a loss-of-coolant accident.
e. For all reactor operating conditions except startup and cold shutdown, the following limiting conditions shall be in effect: (1) One operable diesel-generator power system and one energized 115 kv external line shall be available. If this condition is not met, normal orderly shutdown will be initiated within one hour and the reactor will be in the cold shutdown condition within ten hours.	e. <u>Surveillance for operation with an inoperable diesel-generator</u> – If a diesel-generator becomes inoperable from any cause other than an inoperable support system or preplanned maintenance or testing, within 8 hours, either determine that the cause of the diesel-generator being inoperable does not impact the operability of the operable diesel-generator or demonstrate operability by testing the operable diesel-generator. Operability by testing will be demonstrated by achieving steady state voltage and frequency.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6.5 Radioactive Material Sources

Applicability:

Applies to the limit on source leakage for sealed or start-up sources.

Objective:

To specify the requirements necessary to limit contamination from radioactive source materials.

Specification:

1. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

2. Results of required leak tests performed on sources, if the tests reveal the presence of 0.005 microcurie or more of removable contamination, shall be reported within 90 days.

4.6.5 Radioactive Material Sources

Applicability:

Applies to the periodic testing requirements for source leakage.

Objective:

To assure the capability of each source material container to limit leakage within allowable limits.

Specification:

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except start-up sources subject to core flux, containing radioactive material, other than hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

3.6.11 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs an accident monitoring function.

Objective:

To assure high reliability of the accident monitoring instrumentation.

Specification:

a. During the power operating condition, the accident monitoring instrumentation channels shown in Table 3.6.11-1 shall be operable except as specified in Table 3.6.11-2.

SURVEILLANCE REQUIREMENT

4.6.11 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs an accident monitoring function.

Objective:

To verify the operability of accident monitoring instrumentation.

Specification:

Instrument channels shall be tested and calibrated at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.11.

TABLE 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

	Parameter	Instrument Channel Test	Instrument Channel Calibration
(1)	Deleted		
(2)	Deleted		
(3)	Reactor vessel water level	Note 1	Note 1
(4)	Drywell Pressure Monitor	Note 1	Note 1
(5)	Suppression Chamber Water Level Monitor	Note 1	Note 1
(6)	Deleted		
(7)	Containment High Range Radiation Monitor	Note 1	Note 1
(8)	Suppression Chamber Water Temperature	Note 1	Note 1

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

Applicability:

Applies to the operability of instrumentation that provides protection of the reactor protection system and reactor trip system.

Objective:

To assure the operability of the instrumentation monitoring the power to the reactor protection system and reactor trip system.

Specification:

a. Except as specified in specifications b and c below, two protective relay systems shall be operable for each power supply.

4.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

Applicability:

Applies to the surveillance of instrumentation that provides protection of the reactor protection system and reactor trip system.

Objective:

To verify the operability of protection instrumentation monitoring the power to the reactor protection and reactor trip buses.

Specification:

a. In accordance with the Surveillance Frequency Control Program –
Demonstrate operability of the overvoltage, undervoltage and underfrequency protective instrumentation by performing an instrument channel test. This instrument channel test will consist of simulating abnormal power conditions by applying from a test source, an overvoltage signal, an undervoltage signal and an under-frequency signal to verify that the tripping logic up to but not including the output contactors functions properly.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. With one protective relaying system inoperable, restore the inoperable system to an operable status within 72 hours or remove the power supply from service. c. With both protective relaying systems inoperable, restore at least one to an operable status within 30 minutes or remove the power supply from service.	b. <u>In accordance with the Surveillance Frequency Control Program</u> Demonstrate operability of the overvoltage, undervoltage and underfrequency protective instrumentation by performing an instrument channel test. This instrument channel test will consist of simulating abnormal power conditions by applying from a test source an overvoltage signal, an undervoltage signal and an underfrequency signal to verify that the tripping logic including the output contactors functions properly at least once. In addition, a sensor calibration will be performed to verify the following setpoints. i. Overvoltage $\leq$132 volts, $\leq$4 seconds ii. Undervoltage $\geq$108 volts, $\leq$4 seconds iii. Underfrequency $\geq$ 57 hertz, $\leq$2 seconds

LIMITING CONDITION FOR OPERATION

3.6.13 REMOTE SHUTDOWN PANELS

Applicability:

Applies to the operating status of the remote shutdown panels.

Objective:

To assure the capability of the remote shutdown panels to provide 1) initiation of the emergency condensers independent of the main/auxiliary control room 2) control of the motor-operated steam supply valves independent of the main/auxiliary control room and 3) parameter monitoring outside the control room.

Specification:

a. During power operation, the remote shutdown panels' Functions in Table 3.6.13-1 shall be operable.

SURVEILLANCE REQUIREMENT

4.6.13 REMOTE SHUTDOWN PANELS

Applicability:

Applies to the periodic testing requirements for the remote shutdown panels.

Objective:

To assure the capability of the remote shutdown panels to provide 1) initiation of the emergency condensers independent of the main/auxiliary control room 2) control of the motor-operated steam supply valves independent of the main/auxiliary control room and 3) parameter monitoring outside the control room.

Specification:

The remote shutdown panels surveillance shall be performed as indicated below:

a. Each remote shutdown panel monitoring instrumentation channel shall be demonstrated operable by performance of the operations and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.13-1.

b. In accordance with the Surveillance Frequency Control Program

   1. Each remote shutdown panel shall be demonstrated to initiate the emergency condensers independent of the main/auxiliary control room.

TABLE 4.6.13-1

REMOTE SHUTDOWN PANEL MONITORING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Calibration
Reactor Pressure	Note 1	Note 1[(a)]
Reactor Water Level	Note 1	Note 1[(a)]
Reactor Water Temperature	Note 1	Note 1
Torus Water Temperature	Note 1	Note 1
Drywell Pressure	Note 1	Note 1[(a)]
Emergency Condenser Water Level	Note 1	Note 1
Drywell Temperature	Note 1	Note 1
"All Rods In" Light	Note 1	N/A

(a) The indicator located at the remote shutdown panel will be calibrated at the frequency listed in Table 4.6.13-1. Calibration of the remaining channel instrumentation is provided by Specification 4.6.2.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.13-1.

LIMITING CONDITION FOR OPERATION

3.6.15 MAIN CONDENSER OFFGAS

Applicability:

Applies to the radioactive effluents from the main condenser.

Objective:

To assure that radioactive material is not released to the environment in any uncontrolled manner and is within the limits of 10CFR20 and 10CFR50, Appendix I.

Specification:

The gross radioactivity (beta and/or gamma) rate of noble gases measured at the recombiner discharge shall be limited to less than or equal to 500,000 μCi/sec. This limit can be raised to 1 Ci/sec. for a period not to exceed 60 days provided the offgas treatment system is in operation.

With the gross radioactivity (beta and/or gamma) rate of noble gases at the recombiner discharge exceeding the above limits, restore the gross radioactivity rate to within its limit within 72 hours or be in at least Hot Shutdown within the next 12 hours.

SURVEILLANCE REQUIREMENT

4.6.15 MAIN CONDENSER OFFGAS

Applicability:

Applies to the periodic test and recording requirements of main condenser offgas.

Objective:

To ascertain that radioactive effluents from the main condenser are within allowable values of 10CFR20, Appendix B and 10CFR50, Appendix I.

Specification:

The gross radioactivity (beta and/or gamma) rate of noble gases from the recombiner discharge shall be determined to be within the limits of Specification 3.6.15 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner discharge:

In accordance with the Surveillance Frequency Control Program.

Within 4 hours following an increase on the recombiner discharge monitor of greater than 50%, factoring out increases due to changes in thermal power level and dilution flow changes.

3.7.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN DEMONSTRATIONS

Applicability:

Applies to shutdown margin demonstration in the cold shutdown condition.

Objective:

To assure the capability of the control rod system to control core reactivity.

a. The reactor mode switch may be placed in the startup position to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

   (1) The source range monitors are operable in the noncoincident condition.

   (2) The rod worth minimizer is operable per Specification 3.1.1b(3)(b) and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

4.7.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN DEMONSTRATIONS

Applicability:

Applies to periodic inspections required to perform shutdown margin demonstrations in the cold shutdown condition.

Objective:

To specify the inspections required to perform the shutdown margin demonstration in the cold shutdown condition.

a. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during the performance of a shutdown margin demonstration, verify that:

   (1) The source range monitors are operable per Specification 3.5.1.

   (2) The rod worth minimizer is operable with the required program per Specification 3.1.1b(3)(b) or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown margin demonstration procedure.

6.5.9 Surveillance Frequency Control Program

This program provides controls for the Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of the Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Controlled Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequency," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-63

NINE MILE POINT NUCLEAR STATION, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

1.0 INTRODUCTION

By application dated May12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15134A232)), as supplemented by letters dated October 22 and November 17, 2015 (ADAMS Accession Nos. ML15295A396 and ML15321A253, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station, Unit 1 (NMP1), Technical Specifications (TSs). The requested changes would revise the NMP1 TSs by relocating specific surveillance requirement (SR) frequencies to a licensee-controlled program. The Nuclear Regulatory Commission (NRC) staff published a proposed no significant hazards consideration determination in the *Federal Register* (FR) on January 6, 2016 (81 FR 261).[1]

2.0 REGULATORY EVALUATION

2.1 Description of the Proposed Changes

The licensee proposed to modify the NMP1 TSs by relocating specific SR frequencies to a licensee-controlled program (i.e., the Surveillance Frequency Control Program (SFCP)) in accordance with Nuclear Energy Institute (NEI) 04-10, Revision 1 "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" (ADAMS Accession No. ML071360456). The licensee stated that the proposed change is consistent with the adoption of NRC-approved Technical Specification Task Force (TSTF)

[1] Exelon initially submitted an application to adopt TSTF-425 on March 26, 2015 (ADAMS Accession Nos. ML15089A231 and ML15089A233). That application was noticed in the *Federal Register* on September 29, 2015 (80 FR 58518). However, Exelon stated that its application dated May 12, 2015 superseded the March 26 application (ADAMS Accession No. ML15134A232) in its entirety. Therefore, this safety evaluation concerns the May 12 application alone, which, as stated above, was noticed in the *Federal Register* on January 6, 2016.

Standard Technical Specifications (STS) Change Traveler TSTF-425, Revision 3, "Relocated Surveillance Frequencies to Licensee Control – RITSTF [Risk-Informed TSTF] Initiative 5b (ADAMS Accession No. ML090850642). The FR notice published on July 6, 2009 (74 FR 31996), announced the availability of TSTF-425, Revison 3.

When implemented, TSTF-425, Revision 3, relocates most periodic frequencies of TS surveillances to the SFCP, and provides requirements for the new SFCP in the Administrative Controls section of the TSs. All surveillance frequencies can be relocated except the following:

- Frequencies that reference other approved programs for the specific interval, such as the In-Service Testing Program or the Primary Containment Leakage Rate Testing Program;

- Frequencies that are purely event-driven (e.g., "each time the control rod is withdrawn to the 'full out' position");

- Frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95% RTP" [rated thermal power]); and

- Frequencies that are related to specific conditions (e.g., battery degradation, age and capacity) or conditions for the performance of a surveillance requirement (e.g., "drywell to suppression chamber differential pressure decrease").

The licensee proposed to add the SFCP to TSs, Section 6, "Administrative Controls," Subsection 6.5.9, "Surveillance Frequency Control program." The SFCP describes the requirements for the program to control changes to the relocated surveillance frequencies. The TS Bases for each affected surveillance would be revised to state that the frequency is controlled under the SFCP. The proposed changes to the Administrative Controls section of the TSs to incorporate the SFCP include a specific reference to NEI 04-10, Revision 1, as the basis for making any changes to the surveillance frequencies once they are relocated out of the TSs.

In a letter dated September 19, 2007 (ADAMS Accession No. ML072570267), the NRC staff approved Topical Report NEI 04-10, Revision 1, as acceptable for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, Revision 1, and in the NRC staff's safety evaluation (SE) providing the basis for NRC acceptance of NEI 04-10, Revision 1.

The licensee proposed other changes and deviations from TSTF-425, which are discussed in Section 3.3 of this SE.

2.2 Applicable Commission Policy Statements

In the "Final Policy Statement: Technical Specifications for Nuclear Power Plants," dated July 22, 1993 (58 FR 39132), the NRC addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment or PRA) in STS. In this 1993 publication, the NRC states:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36][2] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed...
>
> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "... probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made ... about the degree of confidence to be given these [probabilistic][3] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety."
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

Approximately two years later the NRC provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities," dated August 16, 1995 (60 FR 42622). In this publication, the NRC states:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach....
>
> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner....

---

[2] This clarification (brackets) is not part of the original policy statement.

[3] The Federal Register Notice 58 FR 39135 (Alteration in Original) explains the brackets.

Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data....

Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:

(1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

(2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

(3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

2.3 Applicable Regulations

In 10 CFR, Section 50.36, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) SRs; (4) design features; and (5) administrative controls. These categories will remain in the NMP1 TSs.

Paragraph 50.36(c)(3) of 10 CFR states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting

conditions for operation will be met." The FR notice published on July 6, 2009 (74 FR 31996), which announced the availability of TSTF-425, Revision 3, states that the addition of the SFCP to the TSs provides the necessary administrative controls to require that surveillance frequencies relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The FR notice also states that changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, Revision 1, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. Such failures can result in the licensee increasing the frequency of a surveillance test. In addition, by having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04-10, Revision 1, the licensee will be required to monitor the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

2.4 Applicable NRC Regulatory Guides

Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML100910006), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML100910008), describes an acceptable risk-informed approach specifically for assessing proposed TS changes.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors (LWRs).

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425, Revision 3, provides for administrative relocation of applicable surveillance frequencies to the SFCP, and provides for the addition of the SFCP to the Administrative Controls section of the TSs. The changes to the Administrative Controls section of the TSs will also require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP. The licensee's application to implement the changes described in TSTF-425, Revision 3, included documentation regarding the PRA technical

adequacy consistent with RG 1.200, Revision 2. NEI 04-10, Revision 1, states that PRA methods are used with plant performance data and other considerations to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is consistent with guidance provided in RG 1.174, Revision 2, and RG 1.177, Revision 1, in support of changes to Surveillance Test Intervals (STIs).

3.1 Review Methodology

RG 1.177, Revision 1, identifies five key safety principles required for risk-informed changes to TSs. Each of these principles is addressed by NEI 04-10, Revision 1.

3.1.1 The Proposed Change Meets Current Regulations

Paragraph 50.36(c)(3) of 10 CFR requires that TSs include surveillances which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The licensee is required by its TSs to perform surveillance tests, calibration, or inspection on specific safety-related equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies are based primarily upon deterministic methods such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved methodologies identified in NEI 04-10, Revision 1, provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

The SRs themselves are remaining in the TSs, as required by 10 CFR 50.36(c)(3). This change is analogous with other NRC-approved TS changes in which the SRs are retained in TSs, but the related surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the In-Service Testing Program and the Primary Containment Leakage Rate Testing Program. Thus, this proposed change complies with 10 CFR 50.36(c)(3) by retaining the requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulatory requirements in 10 CFR 50.65, 10 CFR Part 50, Appendix B, and the monitoring required by NEI 04-10, Revision 1, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied, and that any performance deficiencies will be identified and appropriate corrective actions taken. The licensee's SFCP ensures that SRs specified in the TSs are performed at intervals sufficient to assure the above regulatory requirements are met. Based on the above, the staff concludes that the proposed change meets the first key safety principle of RG 1.177, Revision 1, by complying with current regulations.

3.1.2 The Proposed Change Is Consistent With the Defense-in-Depth Philosophy

The defense-in-depth philosophy (i.e., the second key safety principle of RG 1.177, Revision 1), is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). (Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not impacted.)
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, is maintained.

The changes to the Administrative Controls section of the TSs will require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP. NEI 04-10, Revision 1, uses both the core damage frequency (CDF) and the large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. The guidance of RG 1.174, Revision 2, and RG 1.177, Revision 1, for changes to CDF and LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and common cause failures (CCFs). Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to an increased likelihood of CCFs. Therefore, the NRC staff concludes that both the quantitative risk analysis and the qualitative considerations assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177, Revision 1.

3.1.3 The Proposed Change Maintains Sufficient Safety Margins

The engineering evaluation that will be conducted by the licensee under the SFCP when frequencies are revised will assess the impact of the proposed frequency change to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the proposed surveillance test frequency change is not in conflict with

approved industry codes and standards or adversely affects any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The design, operation, testing methods, and acceptance criteria for SSCs specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plants' licensing bases, including the Updated Final Safety Analysis Report and TS Bases, because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. Based on the above, the NRC staff concludes that safety margins are maintained by the proposed methodology, and the third key safety principle of RG 1.177, Revision 1, is satisfied.

3.1.4 <u>When Proposed Changes Result in an Increase in CDF or Risk, the Increases Should Be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement</u>

RG 1.177, Revision 1, provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies which requires identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. The changes to the Administrative Controls section of the TSs will require application of NEI 04-10, Revision 1, in the SFCP. NEI 04-10, Revision 1, satisfies the intent of RG 1.177, Revision 1, guidance for evaluation of the change in risk, and for assuring that such changes are small by providing the technical methodology to support risk-informed TSs for control of surveillance frequencies.

3.1.4.1 <u>Quality of the PRA</u>

The quality of the licensee's PRA must be commensurate with the safety significance of the proposed TS change and the role the PRA plays in justifying the change. That is, the greater the change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the quality of the PRA.

RG 1.200 provides regulatory guidance for assessing the technical adequacy of a PRA. The current revision (i.e., Revision 2) of this RG endorses, with clarifications and qualifications, the use of (1) American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (i.e., the PRA Standard), (2) NEI 00-02, "PRA Peer Review Process Guidance" (ADAMS Accession Nos. ML061510619 and ML063390593), and (3) NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (ADAMS Accession No. ML083430462).

The licensee has performed an assessment of the NMP1 internal events PRA model, as discussed below, used to support the SFCP using the guidance of RG 1.200 to assure that the PRA model is capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability Category II of the NRC-endorsed PRA standard is the target capability level for supporting requirements for the

internal events PRA for this application. Any identified deficiencies to those requirements are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies where appropriate, in accordance with NEI 04-10, Revision 1.

A full scope peer review was performed in 2008 of the internal events, at-power PRA model. The peer review was based on RG 1.200, Revision 1 and ASME/ANS PRA Standard RA-Sc-2007. It was performed by the Boiling Water Reactors Owners Group, and it followed the industry PRA peer review process, NEI 05-04, and NEI 00-002. The licensee provided the Findings and Observations (F&Os) remaining open in Table 2-1 of the license amendment request (LAR), dated May 12, 2015.

The NRC staff reviewed these internal events PRA F&Os and their dispositions in the LAR to determine whether any gaps in the PRA model were identified that could impact the application. The NRC staff assessed these peer review F&Os to ensure any deficiencies in meeting Capability Category II can be addressed for the SFCP per the NRC-approved NEI 04-10, Revision 1 methodology. The NRC staff found the dispositions to be acceptable because the licensee can acceptably address these F&Os by following NEI 04-10 guidance, including performing appropriate sensitivity analyses and reviews of the PRA model results.

RG 1.200, Revision 2 states: "It is NRC's expectation that, if the results of the self-assessment are used to demonstrate the technical adequacy of a PRA for an application, differences between the current version of the Standard (as endorsed in Appendix A of Revision 2 of this Regulatory Guide), and the earlier version of the ASME PRA Standard (i.e., ASME RA-Sb-2005) be identified and addressed." In supplemental response to PRA RAI 2 in a letter, dated November 17, 2015, the licensee stated that a gap assessment was performed for the internal events PRA between RG 1.200, Revision 1 and RG 1.200, Revision 2. The licensee confirmed that the supporting requirements of the PRA standard ASME RA-Sc-2007, used for the peer review, were not changed from ASME RA-Sb-2005. This gap assessment did not lead to the identification of any new not-met supporting requirements, findings, or observations to the original capability category ranking from the 2008 peer review.

Based on the licensee's assessments using the currently applicable PRA standard and revision of RG 1.200, the NRC staff concludes that the level of PRA quality, combined with the evaluation and disposition of gaps, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 of RG 1.177, Revision 1.

3.1.4.2 Scope of the PRA

The changes to the Administrative Controls section of the TSs will require the licensee to evaluate each proposed change to a relocated surveillance frequency using NEI 04-10, Revision 1, to determine its potential impact on risk (i.e., CDF and LERF) from internal events, fires, seismic, other external events, and shutdown conditions. In cases where a PRA of sufficient scope or quantitative risk models were unavailable, the licensee uses bounding analyses, or other conservative quantitative evaluations. A qualitative screening analysis may be used when the surveillance frequency impact on plant risk is shown to be negligible or zero.

The licensee's PRA model quantifies at-power internal events, including internal flooding. The licensee also has a detailed Fire PRA model which was peer reviewed in 2012 against ASME/ANS PRA Standard RA-Sa-2009 and RG 1.200, Revision 2. The licensee may use these PRA models, in accordance with NEI 04-10, Revision 1, to perform quantitative evaluations to support the development of changes to surveillance frequencies in the SFCP. This is acceptable because the NRC-approved methodology in NEI 04-10, Revision 1, allows for more refined analysis to be performed to support changes to surveillance frequencies in the SFCP.

The licensee does not have PRA modes for external events. The licensee will follow NEI 04-10 guidance for external events hazards which do not have a PRA model. According to the response to PRA RAI 3 in a letter dated October 22, 2015, the primary area of other external event evaluation at NMP1 will be for seismic initiating events. The licensee has evaluated seismic events, high winds, other floods, and other external event hazards for the NMP1 Individual Plant Examination for External Events (IPEEE) study. The IPEEE study had determined high winds, other floods, and other external event hazards to be negligible contributors to the overall plant risk. In order to account for changes in plant conditions since the completion of the IPEEE study, the licensee's qualitative evaluation using NEI 04-10 guidance of these other external events will reflect and consider the current plant configuration and operating experience.

The licensee does not have a shutdown PRA model. According to the response to PRA RAI 4 in the October 22, 2015 letter, the licensee will assess shutdown events qualitatively using NEI 04-10 guidance and their procedural guidance.

Based on the above, the NRC staff concludes that through the application of NRC-approved NEI 04-10, Revision 1, the licensee's evaluation methodology is sufficient to ensure that the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation and is consistent with Regulatory Position 2.3.2 of RG 1.177, Revision 1.

3.1.4.3 PRA Modeling

The licensee's methodology includes the determination of whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted CCF modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to the surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are addressed by the requirements for PRA technical adequacy consistent with guidance contained in RG 1.200, and by sensitivity studies identified in NEI 04-10, Revision 1.

Based on the above, the NRC staff concludes that through the application of NRC-approved NEI 04-10, Revision 1, the NMP1 PRA modeling is sufficient to ensure an acceptable evaluation of risk for the proposed changes in surveillance frequency, and is consistent with Regulatory Position 2.3.3 of RG 1.177, Revision 1.

3.1.4.4 Assumptions for Time Related Failure Contributions

NEI 04-10, Revision 1, criteria adjust the time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency. This is consistent with RG 1.177, Revision 1, Section 2.3.3, which permits separation of the failure rate contributions into demand and standby for evaluation of SRs. According to the PRA RAI 5 response in a letter dated October 22, 2015, the licensee will assess standby time-related failures in accordance with NEI 04-10 by changing the test interval for those SSCs that include a standby periodic tested failure mode in the NMP1 PRA models along with the appropriate adjustments to common cause failure events. If the SSCs do not appear explicitly in the PRA models, then the licensee will perform either a bounding assessment using a surrogate event or a qualitative assessment in accordance with NEI 04-10 guidance. In addition, if there is no standby periodically tested event in the PRA models and one is not added, or the failure cannot be divided into time-based and non-time based (i.e., demand) contributions, the licensee will assume all contributions to the failure rate to be time-based consistent with NEI 04-10 guidance.

The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, such that the failure probability is assumed to increase linearly with time, and will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The NEI 04-10 process requires consideration of qualitative sources of information with regards to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus, the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential benefits of a reduced surveillance frequency, including reduced downtime and reduced potential for restoration errors, test-caused transients, and test-caused wear of equipment, are identified qualitatively, but not quantitatively assessed. Based on the above, the NRC staff concludes that through the application of NRC-approved NEI 04-10, Revision 1, the licensee has employed reasonable assumptions with regard to extensions of STIs, and is consistent with Regulatory Position 2.3.4 of RG 1.177, Revision 1.

3.1.4.5 Sensitivity and Uncertainty Analyses

By having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04 10, Revision 1, the licensee will be required to have sensitivity studies that assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact on the frequency of initiating events, and any identified deviations from Capability Category II of the PRA standard. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. The licensee will also be required to perform monitoring and feedback of SSC performance, once the revised surveillance frequencies are implemented. Based on the above, the NRC staff concludes that through the application of NRC-approved NEI 04-10, Revision 1, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations and is consistent with Regulatory Position 2.3.5 of RG 1.177, Revision 1.

3.1.4.6 Acceptance Guidelines

The licensee will be required to quantitatively evaluate the change in total risk (including internal and external events contributions) in terms of CDF and LERF for both the individual risk impact of a proposed change in surveillance frequency and the cumulative impact from all individual changes to surveillance frequencies using NEI 04-10, Revision 1, in accordance with the TS SFCP. Each individual change to surveillance frequency must show a risk impact below 1E-6 per year for change to CDF and below 1E-7 per year for change to LERF. These changes to CDF and LERF are consistent with the acceptance criteria of RG 1.174, Revision 2, for very small changes in risk. Where the RG 1.174, Revision 2, acceptance criteria are not met, the process in NEI 04-10, Revision 1, either considers revised surveillance frequencies which are consistent with RG 1.174, Revision 2, or the process terminates without permitting the proposed changes. Where quantitative results are unavailable for comparison with the acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or zero. Otherwise, bounding quantitative analyses are required which demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174, Revision 2, acceptance guidelines for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact less than 1E-5 per year for change to CDF, and less than 1E-6 per year for change to LERF, and the total CDF and total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These values are consistent with the acceptance criteria of RG 1.174, Revision 2, as referenced by RG 1.177, Revision 1, for changes to surveillance frequencies.

Consistent with the NRC's SE dated September 19, 2007, for NEI 04-10, Revision 1, the TS SFCP will require the licensee to calculate the total change in risk (i.e., the cumulative risk) by comparing a baseline model that uses failure probabilities based on surveillance frequencies prior to being changed per the SFCP to a revised model that uses failure probabilities based on the changed surveillance frequencies. The NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with insignificant risk increases (i.e., less than 5E-8 CDF and 5E-9 LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance of RG 1.174, Revision 2, is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history. The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. Based on the above, the NRC staff concludes that the licensee's application of NRC-approved NEI 04-10, Revision 1, provides acceptable methods for evaluating the risk increase associated with proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177, Revision 1.

Therefore, the staff concludes that the proposed methodology satisfies the fourth key safety principle of RG 1.177, Revision 1, by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement.

3.1.5 The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies

The licensee's adoption of TSTF-425, Revision 3, requires application of NEI 04-10, Revision 1, in the SFCP. NEI 04-10, Revision 1, requires performance monitoring of SSCs whose surveillance frequencies have been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of Maintenance Rule monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions which may be required by the Maintenance Rule. The performance monitoring and feedback specified in NEI 04-10, Revision 1, is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177, Revision 1. Thus, the staff concludes that the fifth key safety principle of RG 1.177, Revision 1, is satisfied.

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

The licensee proposed including the SFCP and specific requirements into the NMP1 TSs, Section 6.5.9, as follows:

Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure that the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

The proposed program is consistent with the model application of TSTF-425 and, therefore, the staff concludes that it is acceptable.

3.3 Deviations from TSTF-425 and Other Changes

NMP1 has custom TSs that have different formatting, numbering, and language as compared to the STS. This results in differences between the marked-up TSTF-425 pages and the amended

NMP1 TS pages. Allowing NMP1 to apply to relocate their TS surveillance frequencies to a program consistent with the NRC-approved TSTF-425 is consistent with the Commission's Final Policy Statement on TSs as published in the FR on July 22, 1993 (55 FR 39132). The NRC staff has reviewed the licensee's proposed mark-ups and determined that they are consistent with the intent of the NRC-approved TSTF-425. The NRC staff has additionally determined that the proposed changes are consistent with the current formatting and content of the current NMP1 TSs, and therefore, the NMP1 TSs will continue to meet 10 CFR 50.36(c)(3).

There are surveillances included in TSTF-425 that are not included in the NMP1 TSs. TSTF-425 transfers control of frequencies for existing surveillances to the SFCP, but it does not add, delete, or modify the content of the surveillance actions themselves. Based on this, the amendment, which represents a plant specific adoption of TSTF-425, only approves a process for future changes to frequencies for existing surveillances in the NMP1 TSs. Therefore, these administrative differences between the amendment and TSTF-425 are not actual deviations, and they are acceptable and in alignment with NRC-approved TSTF-425.

This amendment transfers control of frequencies for plant specific surveillances (i.e., surveillances not included in TSTF-425) to the SFCP. Although, these changes are not included in the marked-up TS pages for TSTF-425, the TSTF states, "The proposed change relocates all periodic Surveillance Frequencies from the Technical Specifications and places the Frequencies under licensee control in accordance with a new program" and "All surveillances are relocated except...[4 exclusion criteria for the surveillance frequencies are listed]." These statements denote that TSTF-425 applies to all surveillances, including the NMP1 plant specific surveillances, that are periodic and do not meet one of the exclusion criteria. The NRC staff has determined that all of the surveillance frequencies being changed by this amendment are periodic and did not meet the exclusion criteria; therefore, it is acceptable to relocate the surveillance frequencies to the SFCP. These differences between the amendment and TSTF-425 are not actual deviations, and they are acceptable and in alignment with NRC-approved TSTF-425.

3.4 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of specific surveillance frequencies to a licensee-controlled document, and controlling changes to these surveillance frequencies in accordance with a new program, the SFCP, identified in the Administrative Controls of TSs. The SFCP and TSs Section 6.0, Subsection 6.5.9 references NEI 04-10, Revision 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TSs to a licensee-controlled document, provided those frequencies are changed in accordance with the NEI 04-10, Revision 1, which is specified in the Administrative Controls section of the TSs.

The proposed licensee adoption of TSTF-425, Revision 3, and risk-informed methodology of NRC-approved NEI 04-10, Revision 1, as referenced in the Administrative Controls section of the TSs, satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177 and RG 1.174, in that:

- The proposed change meets current regulations;
- The proposed change is consistent with defense-in-depth philosophy;
- The proposed change maintains sufficient safety margins;
- Increases in risk resulting from the proposed change are small and consistent with the Commission's Safety Goal Policy Statement; and
- The impact of the proposed change is monitored with performance measurement strategies.

Section 50.36(c) of 10 CFR discusses the categories that will be included in the TSs. Section 50.36(c)(3) of 10 CFR discusses the specific category of SRs and states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." Based on the above evaluation, the NRC staff concludes that, with the proposed relocation of surveillance frequencies to a licensee-controlled document and administratively controlled in accordance with the TS SFCP, the licensee continues to meet the requirements in 10 CFR 50.36.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and involves changes to SRs. The NRC staff has determined that the amendment involves 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 on January 6, 2016 (81 FR 261) that the amendment involves no significant hazards consideration and there has been no public comment on such finding.

Accordingly, this amendment meets 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 this amendment.

6.0 CONCLUSION

The staff 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) there is reasonable assurance that such activities will be

conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. O'Neal

Date: May 31, 2016

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Brenda L. Mozafari, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures:
1. Amendment No. 222 to DPR-63
2. Safety Evaluation

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DATE	05/18/16	05/18/16	05/18/16	05/18/16	05/18/16
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