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

September 27, 2010

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: RELOCATION OF SURVEILLANCE REQUIREMENT FREQUENCIES TO A LICENSEE CONTROLLED DOCUMENT BASED ON TECHNICAL SPECIFICATION TASK FORCE-425, REVISION 3 (TAC NO. ME2494)

Dear Mr. Pacilio:

The Nuclear Regulatory Commission has issued the enclosed Amendment No.276 to Renewed Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek), in response to your application dated October 30, 2009, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093060126), as supplemented by letter dated April 16, 2010, and August 31, 2010 (ADAMS Accession Nos. ML101060560 and ML102430467, respectively). The amendment revises the Oyster Creek Technical Specifications to relocate a number of Surveillance Requirement frequencies to a licensee controlled document.

A copy of our Safety Evaluation is enclosed and a Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

G. Edward Miller, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures:
1. Amendment No. 276 to Renewed DPR-16
2. Safety Evaluation

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No.276
License No. DPR-16

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

   A. The application for amendment by Exelon Generation Company, LLC, dated October 30, 2009, as supplemented by letters dated April 16, 2010, and August 31, 2010, 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-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 276 , are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements:

   This license amendment is effective as of the date of issuance, and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: September 27, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 276

RENEWED FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following page of the 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	Insert
Page 3	Page 3

Replace the following pages of the 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.

Remove	Insert
i	i
iii	iii
1.0-4	1.0-4
1.0-5	1.0-5
4.1-1	4.1-1
4.1-4	4.1-4
4.1-5	4.1-5
4.1-6	4.1-6
4.1-7	4.1-7
4.1-8	4.1-8
4.1-9	4.1-9
4.1-10	4.1-10
4.2-1	4.2-1
4.2-2	4.2-2
4.3-1	4.3-1
4.3-2	4.3-2
4.4-1	4.4-1
4.4-2	4.4-2
4.5-2	4.5-2
4.5-3	4.5-3
4.5-4	4.5-4
4.5-5	4.5-5
4.5-6	4.5-6
4.5-9	4.5-9

Remove	Insert
4.6-1	4.6-1
4.7-1	4.7-1
4.7-2	4.7-2
4.7-3	4.7-3
4.8-1	4.8-1
4.9-1	4.9-1
4.10-1	4.10-1
4.12-1	4.12-1
4.12-2	4.12-2
4.13-1	4.13-1
4.13-2	4.13-2
4.15-1	4.15-1
4.15-2	4.15-2
4.17-1	4.17-1
--	6-23

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, or special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate such byproduct, source, or special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at steady-state power levels not in excess of 1930 megawatts (thermal) (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 276 are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated March 3, 1978, and supplements thereto, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

*Issued by NRC Order dated 10-24-80

1.19 INSTRUMENTATION SURVEILLANCE DEFINITIONS

A. CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

B. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

C. CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

D. Source Check

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

1.20 FDSAR

Oyster Creek Unit No. 1 Facility Description and Safety Analysis Report as amended by revised pages and figure changes contained in Amendments 14, 31 and 45* and continuing through Amendment 79.

1.21 CORE ALTERATION

A core alteration is the addition, removal, relocation or other manual movement of fuel or controls in the reactor core. Control rod movement with the control rod drive hydraulic system is not defined as a core alteration.

1.22 CRITICAL POWER RATIO

The critical power ratio is the ratio of that power in a fuel assembly which is calculated, by application of an NRC approved CPR correlation, to cause some point in that assembly to experience boiling transition divided by the actual assembly operating power.

1.23 (DELETED)

*Per Erata dtd. 4-9-69

1.24 SURVEILLANCE REQUIREMENTS

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 the safety limits, and that the limiting conditions of operation will be met. Each surveillance requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.[1]

Surveillance requirements for systems and components are applicable only during the modes of operation for which the system or components are required to be operable, unless otherwise stated in the specification.

This definition establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance, e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a fuel cycle length surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for the surveillance that are not performed during refueling outages. The limitation of this definition is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

1.25 APPENDIX J TEST PRESSURE

For the purpose of conducting leak rate tests to meet 10 CFR 50 Appendix J, P_a = 35 psig.

1.26 FRACTION OF LIMITING POWER DENSITY (FLPD)

The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

1.27 MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

[1] For the 10 CFR 50 Appendix J Type A test, the 25% shall not exceed 15 months.

SECTION 4

SURVEILLANCE REQUIREMENTS

4.1 PROTECTIVE INSTRUMENTATION

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

Objective: To specify the minimum frequency and type of surveillance to be applied to the safety instrumentation.

Specification: Instrumentation shall be checked, tested, and calibrated as indicated in Tables 4.1.1 and 4.1.2 using the definitions given in Section 1, and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Tables 4.1.1 and 4.1.2.

TABLE 4.1.1
Page 1 of 6

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel		Check (Note 5)	Calibrate (Note 5)	Test (Note 5)	Remarks (Applies to Test & Calibration)
1.	High Reactor Pressure		Note 3		
2.	High Drywell Pressure (Scram)	N/A			By application of test pressure
3.	Low Reactor Water Level		Note 3		
4.	Low-Low Water Level		Note 3		
5.	High Water Level in Scram Discharge Volume				
	a. Digital	N/A			By varying level in sensor columns
	b. Analog	N/A	Note 3		
6.	Low-Low-Low Water Level	N/A			By application of test pressure
7.	High Flow in Main Steamline				By application of test pressure
8.	Low Pressure in Main Steamline	N/A			By application of test pressure
9.	High Drywell Pressure (Core Cooling)				By application of test pressure
10.	Main Steam Isolation Valve (Scram)	N/A	N/A		By exercising valve

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel		Check (Note 5)	Calibrate (Note 5)	Test (Note 5)	Remarks (Applies to Test & Calibration)
11.	APRM Level	N/A		N/A	Verify the absolute difference between the APRM channels and the calculated power is ≤ 2% rated thermal power [plus any gains required by LSSS 2.3.A.1]
	APRM Scram Trips • Flow based neutron flux – high • Fixed neutron flux – high or inop • Downscale	Note 2			Using built-in calibration equipment during POWER OPERATION
12.	APRM Rod Blocks	Note 2			Upscale and downscale
13.	DELETED				
14.	High Radiation in Reactor Building Operating Floor Ventilation Exhaust				Using gamma source for calibration
15.	High Radiation on Air Ejector Off-Gas				Using built-in calibration equipment Channel Check Source check Calibration according to established station calibration procedures Note a

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel		Check (Note 5)	Calibrate (Note 5)	Test (Note 5)	Remarks (Applies to Test & Calibration)
16.	IRM Level	N/A	Each Startup	N/A	
	IRM Scram	*	*	*	Using built-in calibration equipment
17.	IRM Blocks	N/A	Prior to startup and shutdown	Prior to startup and shutdown	Upscale and downscale
18.	Condenser Low Vacuum	N/A			
19.	Manual Scram Buttons	N/A	N/A		
20.	High Temperature Main Steamline Tunnel	N/A			Using heat source box
21.	SRM	*	*	*	Using built-in calibration equipment
22.	Isolation Condenser High Flow ΔP (Steam & Water)	N/A			By application of test pressure
23.	Turbine Trip Scram	N/A	N/A		
24.	Generator Load Rejection Scram	N/A			
25.	Recirculation Loop Flow	N/A		N/A	By application of test pressure
26.	Low Reactor Pressure Core Spray Valve Permissive	N/A			By application of test pressure

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel		Check (Note 5)	Calibrate (Note 5)	Test (Note 5)	Remarks (Applies to Test & Calibration)
27.	Scram Discharge Volume (Rod Block)				
	a) Water level high	N/A			Calibrate by varying level in sensor column
	b) Scram Trip bypass	N/A	N/A		
28.	Loss of Power				
	a) 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)				
	b) 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)				
29.	Drywell High Radiation	N/A			
30.	Automatic Scram Contactors	N/A	N/A		Note 1
31.	Core Spray Booster Pump Differential Pressure	N/A			By application of a test pressure

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel	Check (Note 5)	Calibrate (Note 5)	Test (Note 5)	Remarks (Applies to Test & Calibration)
32. LPRM Level				
a) Electronics	N/A			
b) Detectors	N/A	Note 4	N/A	
33. RWCU HELB High Temperature	N/A			Perform Channel Tests using the test switches.

* Calibrate prior to startup and normal shutdown and thereafter check and test at the frequency specified in the Surveillance Frequency Control Program until no longer required.

Legend: N/A = Not Applicable

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

NOTE 1: Each automatic scram contactor is required to be tested at the frequency specified in the Surveillance Frequency Control Program. When not tested by other means, the test can be performed by using the subchannel test switches.

NOTE 2: At the frequency specified in the Surveillance Frequency Control Program, the reactor neutron flux peaking factor shall be estimated and flow-referenced APRM scram and rod block settings shall be adjusted, if necessary, as specified in Section 2.3 Specifications A.1 and A.2.

NOTE 3: Calibrate electronic bistable trips by injection of an external test current at the frequency specified in the Surveillance Frequency Control Program. Calibrate transmitters by application of test pressure at the frequency specified in the Surveillance Frequency Control Program.

NOTE 4: Perform LPRM detectors calibration at the frequency specified in the Surveillance Frequency Control Program.

NOTE 5: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

The following notes are only for Item 15 of Table 4.1.1:

A channel may be taken out of service for the purpose of a check, calibration, test or maintenance without declaring the channel to be inoperable.

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

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

TABLE 4.1.2

MINIMUM TEST FREQUENCIES FOR TRIP SYSTEMS

Trip System	Minimum Test Frequency (Note 1)
1) Dual Channel (Scram)	
2) Rod Block	
3) DELETED	DELETED
4) Automatic Depressurization each trip system, one at a time	
5) MSIV Closure each closure logic circuit independently (1 valve at a time)	
6) Core Spray each trip system, one at a time	
7) Primary Containment Isolation each trip circuit independently (1 valve at a time)	
8) Refueling Interlocks	Prior to each refueling operation
9) Isolation Condenser Actuation and Isolation each trip circuit independently (1 valve at a time)	
10) Reactor Building Isolation and SGTS Initiation	
11) DELETED	DELETED
12) Air Ejector Offgass Line Isolation	
13) Containment Vent and Purge Isolation	

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. SDM shall be verified:
   1. Prior to each CORE ALTERATION, and
   2. Once within 4 hours following the first criticality following any CORE ALTERATION.

B. The control rod drive housing support system shall be inspected after reassembly.

C. The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

   1. For all control rods prior to THERMAL POWER exceeding 40% power with reactor coolant pressure greater than 800 psig, following core alterations or after a reactor shutdown that is greater than 120 days.

   2. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either "a" or "b" as follows:

      a.1 Specifically affected individual control rods shall be scram time tested with the reactor depressurized and the scram insertion time from the fully withdrawn position to 90% insertion shall not exceed 2.2 seconds, and

      a.2 Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure prior to exceeding 40% power.

      b. Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure.

   3. At the frequency specified in the Surveillance Frequency Control Program, for at least 20 control rods, on a rotating basis, with reactor coolant pressure greater than 800 psig.

D. Each withdrawn control rod shall be exercised at the frequency specified in the Surveillance Frequency Control Program. This test shall be performed within 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.

E. Surveillance of the standby liquid control system shall be as follows:

1. Pump operability	Note 1
2. Boron concentration determination	Note 1
3. Functional test	Note 1
4. Solution volume and temperature check	Note 1
5. Solution Boron-10 Enrichment	Note 1. Enrichment analyses shall be received no later than 30 days after sampling. If not received within 30 days, notify NRC (within 7 days) of plans to obtain test results.

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted above.

F. At specific power operation conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed with equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

G. The scram discharge volume drain and vent valves shall be verified open at the frequency specified in the Surveillance Frequency Control Program, except in shutdown mode*, and shall be cycled at least one complete cycle of full travel at the frequency specified in the Surveillance Frequency Control Program.

H. All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE. This will be done at the frequency specified in the Surveillance Frequency Control Program by placing the mode switch in shutdown and by verifying that:

a. The drain and vent valves close within 30 seconds after receipt of a signal for control rods to scram, and

b. The scram signal can be reset and the drain and vent valves open when the scram discharge volume trip is bypassed.

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

4.3 REACTOR COOLANT

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification: A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. Pressure and temperature curves are contained in the Pressure and Temperature Limits Report (PTLR).

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a, except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a.

C. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR, Section 50.55a, except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a.

D. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with Specification C above. Setpoints shall be as follows:

Number of Valves	Set Points (psig)
4	1212 ± 36
5	1221 ± 36

F. A sample of reactor coolant shall be analyzed at the frequency specified in the Surveillance Frequency Control Program for the purpose of determining the content of chloride ion and to check the conductivity.

* G. Primary Coolant System Pressure Isolation Valves Specification:

1. Periodic leakage testing[a] on each valve listed in Table 4.3.1 shall be accomplished prior to exceeding 600 psig reactor pressure every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, whenever the valve is moved whether by manual actuation or due to flow conditions, and after returning the valve to service after maintenance, repair or replacement work is performed.

H. Reactor Coolant System Leakage

1. Unidentified leakage rate shall be calculated at the frequency specified in the Surveillance Frequency Control Program.

2. Total leakage rate (identified and unidentified) shall be calculated at the frequency specified in the Surveillance Frequency Control Program.

3. A CHANNEL CALIBRATION of the primary containment sump flow integrator and the primary containment equipment drain tank flow integrator shall be conducted at the frequency specified in the Surveillance Frequency Control Program.

I. An inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in the generic letter or in accordance with alternate measures approved by the NRC staff.

Bases:

Data is available relating neutron fluence (E>1.0MeV) and the change in the Reference Nil-Ductility Transition Temperature (RT_{NDT}). Pressure and temperature curves are contained in the Pressure and Temperature Limits Report (PTLR).

The inspection program will reveal problem areas should they occur, before a leak develops. In addition, extensive visual inspection for leaks will be made on critical systems. Oyster Creek was designed and constructed prior to

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

* NRC Order dated April 20, 1981.

4.4 EMERGENCY COOLING

Applicability: Applies to surveillance requirements for the emergency cooling systems.

Objective: To verify the operability of the emergency cooling systems.

Specification: Surveillance of the emergency cooling systems shall be performed as follows:

Item	Frequency
A. Core Spray System	
1. Pump Operability	Note 1. Also after major maintenance and prior to startup following a refueling outage.
2. Motor operated valve operability	Note 1
3. Automatic actuation test	Note 1
4. Pump compartment water-tight doors closed	Note 1 and after each entry.
5. Core spray header ΔP instrumentation	
CHANNEL CHECK	Note 1
CHANNEL CALIBRATION	Note 1
CHANNEL FUNCTIONAL TEST	Note 1
B. Automatic Depressurization	
1. Verify each relief valve actuator strokes when manually actuated	Note 1
2. Automatic actuation test	Note 1
C. Containment Cooling System	
1. Pump Operability	Note 1. Also after major maintenance and prior to startup following a refueling outage.

Item	Frequency
C. Containment Cooling System	
2. Motor-operated valve operability	Note 1
3. Pump compartment water-tight doors closed	Note 1 and after each entry.
D. Emergency Service Water System	
1. Pump Operability	Note 1. Also after major maintenance and prior to startup following a refueling outage.
E. Control Rod Drive Hydraulic System	
1. Pump Operability	Note 1. Also after major maintenance and prior to startup following a refueling outage.
F. Fire Protection System	
1. Pump Operability	Note 1. Also after major maintenance and prior to startup following a refueling outage.
2. Isolation valve operability	Note 1. Also after major maintenance and prior to startup following a refueling outage.

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted.

Bases:

It is during major maintenance or repair that a system's design intent may be violated accidentally. Therefore, a functional test is required after every major maintenance operation. During an extended outage, such as a refueling outage, major repair and maintenance may be performed on many systems. To be sure that these repairs on other systems do not encroach unintentionally on critical standby cooling systems, they should be given a functional test prior to startup.

Motor operated pumps, valves and other active devices that are normally on standby should be exercised periodically to make sure that they are free to operate. Motors on pumps should operate long enough to approach equilibrium temperature to ensure there is no overheat problem. Whenever practical, valves should be stroked full length to ensure that nothing impedes their motion. Testing of components per OC Inservice Testing Program in accordance with the ASME Code provides assurances of the availability of the system. The Control Rod Hydraulic pumps and Fire Protection System pumps are not part of the Inservice Test Program per the ASME Code and will continue to be tested for operability at the frequency specified in the Surveillance Frequency Control Program.

b. If the airlock is opened during a period when Primary Containment is not required, it need not be tested while Primary Containment is not required, but must be tested at P_a prior to returning the reactor to an operating mode requiring PRIMARY CONTAINMENT INTEGRITY.

D. Primary Containment Leakage Rates shall be limited to:

1. The maximum allowable Primary Containment leakage rate is 1.0 L_a. The maximum allowable Primary Containment leakage rate to allow for plant startup following a type A test is 0.75 L_a. The leakage rate acceptance criteria for the Primary Containment Leakage Rate Testing Program for Type B and Type C tests is $\leq$0.60 L_a at P_a, except as stated in Specification 4.5.D.2.

2. Verify leakage rate through each MSIV is $\leq$ 11.9 scfh when tested at $\geq$ 20 psig.

3. The leakage rate acceptance criteria for the drywell airlock shall be $\leq$ 0.05 L_a when measured or adjusted to P_a.

E. Continuous Leak Rate Monitor

1. When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements.

2. This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

F. Functional Test of Valves

1. All automatic primary containment isolation valves shall be tested for automatic closure by an isolation signal at the frequency specified in the Surveillance Frequency Control Program and the isolation time determined to be within its limit. The following valves are required to close in the time specified below:

   Main steam line isolation valves: $\geq$ 3 seconds and $\leq$ 10 seconds

2. Each automatic primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on

the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel and verifying the isolation time limit is met. Following maintenance, repair or replacement work on the control or power circuit for the valves, the affected component shall be tested to assure it will perform its intended function in the circuit.

3. During each COLD SHUTDOWN, each main steam isolation valve shall be closed and its closure time verified to be within the limits of Specification 4.5.F.1 above unless this test has been performed within the last 92 days.

4. Reactor Building to Suppression Chamber Vacuum Breakers

   a. The reactor building to suppression chamber vacuum breakers and associated instrumentation, including setpoint, shall be checked for proper operation at the frequency specified in the Surveillance Frequency Control Program.

   b. At the frequency specified in the Surveillance Frequency Control Program, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker from closed to fully open does not exceed the force specified in Specification 3.5.A.4.a. The air-operated vacuum breaker instrumentation shall be calibrated at the frequency specified in the Surveillance Frequency Control Program.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic OPERABILITY Tests

      At the frequency specified in the Surveillance Frequency Control Program and following any release of energy which would tend to increase pressure to the suppression chamber, each OPERABLE suppression chamber - drywell vacuum breaker shall be exercised. Operation of position switches, indicators and alarms shall be verified at the frequency specified in the Surveillance Frequency Control Program by operation of each OPERABLE vacuum breaker.

   b. The following tests, with the exception of b(4), are performed at the frequency specified in the Surveillance Frequency Control Program.

      (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) The suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

(3) At least four of the suppression chamber - drywell vacuum breakers shall be inspected. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected such that Specification 3.5.A.5.a can be met.

(4) A drywell to suppression chamber leak rate test shall be performed once every 24 months to 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 air flow through a 2-inch orifice.

G. Reactor Building

1. Secondary containment capability tests shall be conducted after isolating the reactor building and placing either Standby Gas Treatment System filter train in operation.

2. The tests shall be performed at the frequency specified in the Surveillance Frequency Control Program and shall demonstrate the capability to maintain a 1/4 inch of water vacuum under calm wind conditions with a Standby Gas Treatment System Filter train flow rate of not more than 4000cfm.

3. A secondary containment capability test shall be conducted at each refueling outage prior to refueling.

4. The results of the secondary containment capability tests shall be in the subject of a summary technical report which can be included in the reports specified in Section 6.

H. Standby Gas Treatment System

1. The capability of each Standby Gas Treatment System circuit shall be demonstrated by:

   a. At the frequency specified in the Surveillance Frequency Control Program, after every 720 hours of operation, and following significant painting, fire, or chemical release in the reactor building during operation of the Standby Gas Treatment System by verifying that:

      (1) The charcoal absorbers remove ≥99% of a halogenated hydrocarbon refrigerant test gas and the HEPA filters remove ≥99% of the DOP in a cold DOP test when tested in accordance with ANSI N510-1975.

(2) Results of laboratory carbon sample analysis show ≥95% radioactive methyl iodide removal efficiency when tested in accordance with ASTM D 3803-1989 (30°C, 95% relative humidity, at least 45.72 feet per minute charcoal bed face velocity).

b. At the frequency specified in the Surveillance Frequency Control Program by demonstrating:

(1) That the pressure drop across a HEPA filter is equal to or less than the maximum allowable pressure drop indicated in Figure 4.5.1.

(2) The inlet heater is capable of at least 10.9 KW input.

(3) Operation with a total flow within 10% of design flow.

c. At the frequency specified in the Surveillance Frequency Control Program by operating each circuit for a minimum of 10 hours.

d. Anytime the HEPA filter bank or the charcoal absorbers have been partially or completely replaced, the test per 4.5.H.1.a (as applicable) will be performed prior to returning the system to OPERABLE STATUS.

e. Automatic initiation of each circuit at the frequency specified in the Surveillance Frequency Control Progam.

I. Inerting Surveillance

When an inert atmosphere is required in the primary containment, the oxygen concentration in the primary containment shall be checked at the frequency specified in the Surveillance Frequency Control Program.

J. Drywell Coating Surveillance

Carbon steel test panels coated with Firebar D shall be placed inside the drywell near the reactor core midplane level. They shall be removed for visual observation and weight loss measurements during the first, second, fourth and eighth refueling outages.

K. Instrument Line Flow Check Valves Surveillance

The capability of a representative sample of instrument line flow check valves to isolate shall be tested at the frequency specified in the Surveillance Frequency Control Program. In addition, each time an instrument line is returned to service after any condition which could have produced a pressure flow disturbance in that line, the open position of the flow check valve in that line shall be verified. Such conditions include:

Leakage at instrument fittings and valves
Venting an unisolated instrument or instrument line
Flushing or draining an instrument
Installation of a new instrument or instrument line

L. Suppression Chamber Surveillance

1. At the frequency specified in the Surveillance Frequency Control Program, the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

2. A visual inspection of the suppression chamber interior, including water line regions, shall be made at the frequency specified in the Surveillance Frequency Control Program.

3. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed until the heat addition is terminated.

4. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160°F or above while the reactor primary coolant system pressure is greater than 180 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.

M. Shock Suppressors (Snubbers)

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

1. Each snubber shall be demonstrated OPERABLE by performance of the following inspection program:

   a. Visual Inspections

      Snubbers are categorized as inaccessible or accessible during reactor operation. Each of the categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.5-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.5-1.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 24 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records. Service life shall not at any time affect reactor operations.

N. Secondary Containment Isolation Valves

1. Each secondary containment isolation valve shall be demonstrated operable prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel. Following maintenance, repair or replacement work on the control or power circuit for the valves, the affected component shall be tested to assure it will perform its intended function in the circuit.

2. At the frequency specified in the Surveillance Frequency Control Program, all valves shall be tested for automatic closure by an isolation signal.

4.6 RADIOACTIVE EFFLUENT

Applicability: Applies to monitoring of gaseous and liquid radioactive effluents of the Station during release of effluents via the monitored pathway(s). Each Surveillance Requirement applies whenever the corresponding Specification is applicable unless otherwise stated in an individual Surveillance Requirement. Surveillance Requirements do not have to be performed on inoperable equipment.

Objective: To measure radioactive effluents adequately to verify that radioactive effluents are as low as is reasonable achievable and within the limit of 10 CFR Part 20.

Specification:

A. Reactor Coolant

Reactor coolant shall be sampled and analyzed at the frequency specified in the Surveillance Frequency Control Program for DOSE EQUIVALENT I-131 during RUN MODE, STARTUP MODE and SHUTDOWN CONDITION.

B. NOT USED.

C. Radioactive Liquid Storage

1. Liquids contained in the following tanks shall be sampled and analyzed for radioactivity at the frequency specified in the Surveillance Frequency Control Program when radioactive liquid is being added to the tank:

   a. Waste Surge Tank, HP-T-3;
   b. Condensate Storage Tank.

D. Main Condenser Offgas Treatment

RELOCATED TO THE ODCM.

E. Main Condenser Offgas Radioactivity

1. The gross radioactivity in fission gases discharged from the main condenser air ejector shall be measured by sampling and analyzing the gases.

   a. At the frequency specified in the Surveillance Frequency Control Program, and

   b. When the reactor is operating at more than 40 percent of rated power, within 4 hours after an increase in the fission gas release via the air ejector of more than 50 percent, as indicated by the Condenser Air Ejector Offgas Radioactivity Monitor after factoring out increase(s) due to change(s) in the THERMAL POWER level.

F. Condenser Offgas Hydrogen Concentration

The concentration of hydrogen in offgases downstream of the recombiner in the Offgas System shall be monitored with hydrogen instrumentation as described in Table 3.15.2.

G. NOT USED.

H. NOT USED.

4.7 AUXILIARY ELECTRICAL POWER

Applicability: Applies to surveillance requirements of the auxiliary electrical supply.

Objective: To verify the availability of the auxiliary electrical supply.

Specification:

A. Diesel Generator

1. Each diesel generator shall be started and loaded to not less than 80% rated load at the frequency specified in the Surveillance Frequency Control Program.

2. The two diesel generators shall be automatically actuated and functionally tested at the frequency specified in the Surveillance Frequency Control Program. This shall include testing of the diesel generator load sequence timers listed in Table 3.1.1.

3. Deleted.

4. The diesel generators' fuel supply shall be checked following the above tests.

5. The diesel generators' starting batteries shall be tested and monitored per Specification 4.7.B.

B. Diesel Generator Starting Batteries

1. Surveillance will be performed at the frequency specified in the Surveillance Frequency Control Program to verify the following:

    a. The active metallic surface of the plates shall be fully covered with electrolyte in all batteries.

    b. The designated pilot cell voltage is greater than or equal to 2.0 volts.

    c. The overall battery voltage is greater than or equal to 112 volts while the battery is on a float charge.

    d. The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.190.

2. Surveillance will be performed at the frequency specified in the Surveillance Frequency Control Program to verify the specific gravity for each fourth cell is greater than or equal to 1.190 when corrected to 77°F. The specific gravity and electrolyte temperature of every fourth cell shall be recorded for surveillance review.

3. Surveillance will be performed at the frequency specified in the Surveillance Frequency Control Program to verify the specific gravity for each cell is greater than or equal to 1.190 when corrected to 77°F. The electrolyte temperature and specific gravity for every cell shall be recorded for surveillance review.

4. At the frequency specified in the Surveillance Frequency Control Program, the diesel generator battery capacity shall be demonstrated to be able to supply the design duty loads (diesel start) during a battery service test.

5. At the frequency specified in the Surveillance Frequency Control Program, the following tests will be performed:

   a. Battery capacity shall be demonstrated to be at least 80% of the manufacturers' rating when subjected to a battery capacity discharge test.

   b. If a Diesel Generator Starting Battery is demonstrated to have less than 85% of manufacturers ratings during a capacity discharge test, it shall be replaced within 2 years.

C. Station Batteries

1. Surveillance will be performed at the frequency specified in the Surveillance Frequency Control Program to verify the following:

   a. The overall battery voltage is greater than or equal to the minimum established float voltage.

   b. Each station battery float current is ≤ 2 amps when battery terminal voltage is greater than or equal to the minimum established float voltage of 4.7.C.1.a.

2. Surveillance will be performed at the frequency specified in the Surveillance Frequency Control Program to verify the following:

   a. The electrolyte level in each station battery is greater than or equal to minimum established design limits.

   b. The voltage of each pilot cell is greater than or equal to 2.07 volts while the respective battery is on a float charge.

   c. The electrolyte temperature of each station battery pilot cell is greater than or equal to minimum established design limits.

3. Surveillance will be performed at the frequency specified in the Surveilllance Frequency Control Program to verify the voltage of each connected cell is greater than or equal to 2.07 volts while the respective battery is on a float charge.

4. At the frequency specified in the Surveillance Frequency Control Program:

   a. The station battery capacity shall be demonstrated to be able to supply the design duty cycle loads during a battery service test. The modified performance discharge test may be substituted for the service test.

   b. (i) Verify required station battery charger supplies ≥ 429 amps for the B MG Set charger, ≥ 600 amps for the A/B static charger, and ≥ 500 amps for the C charger, for ≥ 4 hours at greater than or equal to the minimum established float voltage, or

      (ii) Verify each required battery charger can recharge the battery to the fully charged state while supplying the normal steady state DC loads during station operation, after a battery discharge to the bounding design basis event discharge state.

5. The following tests will be performed to verify battery capacity:

   a. At the frequency specified in the Surveillance Frequency Control Progam, battery capacity shall be demonstrated to be at least 80% of the manufacturers' rating when subjected to a performance discharge test or a modified performance discharge test.

   b. Performance discharge tests or modified performance discharge tests of station battery capacity shall be given at least once per 12 months when:

      (i) The station battery shows degradation, or

      (ii) The station battery has reached 85% of expected life with battery capacity < 100% of manufacturer's rating.

   c. Performance discharge tests or modified performance discharge tests of station battery capacity shall be given at least once per 24 months when the battery has reached 85% of expected life with battery capacity ≥ 100% of manufacturer's rating.

4.8 ISOLATION CONDENSER

Applicability: Applies to periodic testing requirements for the isolation condenser system.

Objective: To verify the operability of the isolation condenser system.

Specification: A. Surveillance of each isolation condenser loop shall be as follows:

Item	Frequency
1. Operability of motor-operated isolation valves and condensate makeup valves.	Note 1
2. Automatic actuation and functional test.	Note 1 or following major repair.
3. Shell side water volume check	Note 1
4. Isolation valve (steam side)	
a. Visual inspection	Note 1
b. External leakage check	Each primary system Leak test
c. Area temperature check	Note 1

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted above.

Basis: Motor-operated valves on the isolation condenser steam and condensate lines and on the condensate makeup line that are normally on standby should be exercised periodically to make sure that they are free to operate. The valves will be stroked full length every time they are tested to verify proper functional performance. This frequency of testing is consistent with instrumentation tests discussed in Specification 4.1. Testing of these components per the ASME Code at the frequency specified in the Surveillance Frequency Control Program provides assurance of availability of the system. Also, at this frequency of testing, wearout should not be a problem throughout the life of the plant.

The automatic actuation and functional test will demonstrate the automatic opening of the condensate return line valves and the automatic closing of the isolation valves on the vent lines to the main steam lines. Automatic closure of the isolation condenser steam and condensate lines on actuation of the condenser pipe break detectors will also be verified by the test. It is during a major maintenance or repair that a system's design intent may be violated accidentally. This makes the functional test necessary after every major repair operation.

By virtue of normal plant operation the operators daily observe the water level in the isolation condensers. In addition, isolation condenser shell side water level sensors provide control room annunciation of condenser high or low water level.

4.9 REFUELING

Applicability: Applies to the periodic testing of those interlocks and instruments used during refueling.

Objective: To verify the operability of instrumentation and interlocks in use during refueling.

Specification:

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

B. Prior to beginning any core alterations, the source range monitors (SRMs) shall be calibrated. Thereafter, the SRM's will be checked, tested, and calibrated at the frequencies specified in the Surveillance Frequency Control Program until no longer required.

C. Within four (4) hours prior to the start of control rod removal pursuant to Specification 3.9.E verify:

   1. That the reactor mode switch is locked in the refuel position and that the one rod out refueling interlock is operable.

   2. That two (2) SRM channels, one in the core quadrant where the control rod is being removed and one in an adjacent quadrant, are operable and inserted to the normal operation level.

D. Verify within four (4) hours prior to the start of control rod removal pursuant to Specification 3.9.F and at the frequency specified in the Surveillance Frequency Control Program thereafter, until replacement of all control rods or rod drive mechanisms and all control rods are fully inserted that:

   1. the reactor mode switch is locked in the refuel position and the one rod out refueling interlock is operable.

   2. Two (2) SRM channels, one in the core quadrant where a control rod is being removed and one in an adjacent quadrant, are operable and fully inserted.

   3. All control rods not removed are fully inserted with the exception of one rod which may be partially withdrawn not more than two notches to perform refueling interlock surveillance.

   4. The four fuel assemblies surrounding each control rod or rod drive mechanism being removed or maintained at the same time are removed from the core cell.

4.10 ECCS RELATED CORE LIMITS

Applicability: Applies to the periodic measurement during power operation of core parameters related to ECCS performance.

Objective: To assure that the limits of Section 3.10 are not being violated.

Specification:

A. Average Planar LHGR.

The APLHGR for each type of fuel as a function of average planar exposure shall be checked at the frequency specified in the Surveillance Frequency Control Program during reactor operation at greater than or equal to 25% rated thermal power.

B. Local LHGR.

The LHGR as a function of core height shall be checked at the frequency specified in the Surveillance Frequency Control Program during reactor operation at greater than or equal to 25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR).

1. MCPR shall be checked at the frequency specified in the Surveillance Frequency Control Program during reactor operation at greater than or equal to 25% rated thermal power.

2. The MCPR operating limit shall be determined within 72 hours of completing scram time testing as required in Specification 4.2.C.

Bases:

The LHGR shall be periodically checked at the frequency specified in the Surveillance Frequency Control Program to determine whether fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, a periodic check of power distribution is adequate.

The minimum critical power ratio (MCPR) is unlikely to change significantly during steady state power operation. In the event of a single pump trip, the surveillance frequency specified in the Surveillance Frequency Control Program remains acceptable because the accompanying power reduction is much larger than the change in MAPLHGR limits for four loop operation at the corresponding lower steady state power level as compared to five loop operation. The frequency specified in the Surveillance Frequency Control Program is also acceptable for the APRM status check since neutron monitoring system failures are infrequent and a downscale failure of an APRM initiates a control rod withdrawal block, thus precluding the possibility of a control rod withdrawal error.

4.12 Alternate Shutdown Monitoring Instrumentation

Applicability: Applies to the surveillance requirements of the alternate shutdown monitoring instrumentation.

Objective: To specify the minimum frequency and type of surveillance to be applied to the alternate shutdown monitoring instrumentation.

Specification:

Each of the alternate shutdown monitoring channels shown in Table 4.12-1 shall be demonstrated operable by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.12-1.

Basis:

The operability of the alternate shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of hot shutdown of the plant from locations outside of the control room. The surveillance frequencies are controlled under the Surveillance Frequency Control Program.

TABLE 4.12-1 ALTERNATE SHUTDOWN

MONITORING INSTRUMENTATION

Functional Limit	CHANNEL CHECK (Note 1)	CHANNEL CALIBRATION (Note 1)
Reactor Pressure		
Reactor Water Level (fuel zone)	n/a	
Condensate Storage Tank Level		
Service Water Pump Discharge Pressure		
Control Rod Drive System Flowmeter		
Shutdown Cooling System Flowmeter	n/a	
Isolation Condenser "B" Shell Water Level		
Reactor Building Closed Cooling Water Pump Discharge Pressure		

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

4.13 ACCIDENT MONITORING INSTRUMENTATION

Applicability: Applies to surveillance requirements for the accident monitoring instrumentation.

Objective: To verify the operability of the accident monitoring instrumentation.

Specification:

A. Safety & Relief Valve Position Indicators

Each primary and safety valve position indicator (primary detector*), relief and safety valve position indicator (backup indications**), and relief valve position indicator (common header temperature element**) channel shall be demonstrated operable by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

B. Wide Range Drywell Pressure Monitor

Each wide range drywell pressure monitor (PT/PR 53 & 54) channel shall be demonstrated operable by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

C. Wide Range Torus Water Level Monitor

Each wide range torus water level monitor (LT/LR 37 & 38) channel shall be demonstrated operable by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

D. DELETED

E. Containment High-Range Radiation Monitor

Each containment high range radiation monitor channel shall be demonstrated operable by performance of the CHANNEL CHECK and CHANNEL CALIBRATION*** operations at the frequencies specified in the Surveillance Frequency Control Program.

F. High Range Radioactive Noble Gas Effluent Monitor

Each high range radioactive noble gas effluent monitor (main stack and turbine building vent) channel shall be demonstrated operable by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

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* Acoustic Monitor

** Thermocouple

*** CHANNEL CALIBRATION for the containment high range radiation monitor shall consist of electronic signal substitution of the channel, not including the detector, for all decades above 10R/hr and a one point calibration check of the detector at or below 10R/hr by means of a calibrated portable radiation source traceable to NBS.

INFORMATION ON THIS PAGE HAS BEEN DELETED

4.15 Explosive Gas Monitoring Instrumentation

Applicability: States surveillance requirements for OPERABILITY of explosive gas monitoring instrumentation.

Objective: To demonstrate the OPERABILITY of explosive gas monitoring instrumentation.

Specification:

Gaseous Effluent Instrumentation

Each explosive gas effluent monitoring instrument channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.15.2.

TABLE 4.15.2

EXPLOSIVE GAS MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK (h)	SOURCE CHECK	CHANNEL CALIBRATION(f)(h)	CHANNEL FUNCTIONAL TEST (h)	CHANNEL SURVEILLANCE REQUIRED (a)
1. Main Condenser Offgas Treatment System Hydrogen Monitor		N/A	(g)		(c)

Legend: N/A = Not Applicable.

TABLE 4.15.2 NOTATIONS

(a) Instrumentation shall be OPERABLE and in service except that a channel may be taken out of services for the purpose of a check, calibration, test or maintenance without declaring it to be inoperable.

(c) During main condenser offgas treatment system operation.

(f) The CHANNEL CALIBRATION shall be performed according to established station calibration procedures.

(g) A CHANNEL CALIBRATION shall include the use of at least two standard gas samples, each containing a known volume percent hydrogen in the range of the instrument, balance nitrogen.

(h) Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

4.17 Control Room Heating, Ventilating, and Air-Conditioning System

Applicability: Applies to surveillance requirements for the control room heating, ventilating, and air conditioning (HVAC) systems.

Objective: To verify the capability of each control room HVAC system to minimize the amount of radioactivity from entering the control room in the event of an accident.

Specification: Surveillance of each control room HVAC system shall be as follows:

A. At the frequency specified in the Surveillance Frequency Control Program: by initiating, from the control room, the partial recirculation mode of operation, and by verifying that the system components are aligned such that the system is operating in this mode.

B. At the frequency specified in the Surveillance Frequency Control Program: by verifying that in the partial recirculation mode of operation, the control room and lower cable spreading room are maintained at a positive pressure of ≥ 1/8 in. WG relative to the outside atmosphere.

Basis: Periodic surveillance of each control room HVAC system is required to ensure the operability of the system. The operability of the system in conjunction with control room design provisions is based upon limiting the radiation exposure to personnel occupying the control room to less than a 30-day integrated dose of 5 rem TEDE for the most limiting design basis accident.

6.24 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 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 Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Definition 1.24 and Surveillance Requirement 4.0.2 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. 276 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-16

EXELON GENERATION COMPANY, LLC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated October 30, 2009 (Reference 9), as supplemented by letters dated April 16, 2010 (Reference 10), and August 31, 2010 (Reference 11), Exelon Generation Company, LLC (Exelon, the licensee) proposed changes to the Technical Specifications (TSs) for Oyster Creek Nuclear Generating Station (OCNGS). The supplemental letter provided clarifying information that did not change the scope of the amendment as described in the initial notice of the proposed action published in the *Federal Register* on December 29, 2009 (74 FR 68869), or the U.S. Nuclear Regulatory Commission (NRC) staff's proposed no significant hazards consideration determination.

The requested change is the adoption of NRC-approved Technical Specification Task Force (TSTF) 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control—RITSTF [Risk-Informed TSTF] Initiative 5b" (Reference 1). When implemented, TSTF-425 relocates most periodic frequencies of TS surveillances to a licensee controlled program, the Surveillance Frequency Control Program (SFCP), and provides requirements for the new program in the Administrative Controls section of the TS. All surveillance frequencies can be relocated except:

- 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"); 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").

A new program is added to the Administrative Controls of TS Section 6 as Specification 6.24. The new program is called the SFCP and describes the requirements for the program to control changes to the relocated surveillance frequencies. The proposed changes to the Administrative Controls of the TS to incorporate the SFCP include a specific reference to Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1 (Reference 2) as the basis for making any changes to the surveillance frequencies once they are relocated out of the TS.

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

2.0 REGULATORY EVALUATION

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

> 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 10 CFR 50.36] 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] 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 2 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" published in the *Federal Register* (60 FR 42622, August 16, 1995). In this FR publication, the NRC states, in part:

> 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.
>
> 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 Title 10 of the Code of Federal Regulations (10 CFR) 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process 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."

In Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical Specifications," the NRC established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TS 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 (LCOs); (3) SRs; (4) design features; and (5) administrative controls.

As stated in 10 CFR 50.36(c)(3), "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." To meet this requirement, the SR must specify both an adequate test, calibration, or inspection and appropriate frequency for performance. Exelon has proposed to implement changes to surveillance frequencies in the SFCP using the methodology contained in NEI 04-10, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and required to be documented. Furthermore, changes to frequencies are subject to regulatory review and oversight of the SFCP implementation through the rigorous NRC review of safety-related SSC performance provided by the reactor oversight program (ROP).

Licensees are required by TS to perform surveillance test, calibration, or inspection on specific safety-related system equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies, currently identified in TS, 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 provides a way to establish risk-informed surveillance frequencies that complement the deterministic approach and support the NRC's traditional defense-in-depth philosophy.

The licensee's SFCP is intended to ensure that SRs specified in the TS are required to be performed at intervals sufficient to assure the above regulatory requirements are met. Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and 10 CFR 50 Appendix B (corrective action program), require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. One of these actions may be to consider increasing the frequency at which a surveillance test is performed. In addition, the SFCP implementation guidance in NEI 04-10 requires monitoring the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs. These requirements, and the monitoring required by NEI 04-10, are intended to 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.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 5), describes a risk-informed approach, acceptable to the NRC, 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 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 4), 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.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425 for OCNGS provides for relocation and administrative control of applicable surveillance frequencies, and provides for the addition of the SFCP to the administrative controls of TS. TSTF-425 also requires the application of NEI 04-10 for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes proposed in TSTF-425 included documentation regarding the PRA technical adequacy consistent with the requirements of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 (Reference 4). In accordance with NEI 04-10 PRA, methods are used, in combination with plant performance data and other considerations, to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is in accordance with guidance provided in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 5) and RG 1.177 in support of changes to surveillance test intervals.

3.1 RG 1.177 Five Key Safety Principles

RG 1.177 identifies five key safety principles required for risk-informed changes to TS. Each of these principles is addressed by the industry methodology document, NEI 04-10.

3.1.1 The Proposed Change Meets Current Regulations

10 CFR 50.36(c)(3) provides that TSs will 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." NEI 04-10 provides guidance for relocating the surveillance frequencies from the TSs to a licensee-controlled program by providing an NRC-approved methodology for

control of the surveillance frequencies. The surveillances themselves would remain in the TSs, as required by 10 CFR 50.36(c)(3).

Exelon has proposed to add Section 6.24 of the TSs which requires any changes to the SR frequencies to be made in accordance with NEI 04-10, Revision 1. The NRC staff has previously found NEI 04-10, Revision 1, to be an acceptable control program for this type of application (Reference 12). Thus, this proposed change meets the first key safety principle of RG 1.177 by complying with current regulations.

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

Consistency with the defense-in-depth philosophy, the second key safety principle of RG 1.177, 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 proposed TS Section 6.24 would require the application of NEI 04-10 for any changes to surveillance frequencies within the SFCP. NEI 04–10 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 and RG 1.177 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. 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 common cause failures. 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.

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 with the principle 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 plant licensing basis (including the Updated Final Safety Analysis Report and bases to TS), since 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.

Thus, safety margins are maintained by the proposed methodology, and the third key safety principle of RG 1.177 is satisfied.

3.1.4 When Proposed Changes Result in an Increase in Core Damage Frequency or Risk, the Increases Should Be Small and Consistent With the Intent of the Commission's Safety Goal Policy Statement

RG 1.177 provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies. This requires the 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 proposed TS 6.24 would require application of NEI 04-10 in the SFCP. As discussed before, NEI 04-10 has previously been found by the NRC staff to satisfy the intent of RG 1.177 requirements for evaluating the change in risk, and for assuring that such changes are small.

3.1.4.1 Quality of the PRA

The quality of the OCNGS PRA is compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. That is, the more the potential 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.

The licensee used RG 1.200 to address the technical adequacy of the OCNGS PRA. RG 1.200 is NRC's developed regulatory guidance, which endorses with comments and qualifications the use of the American Society of Mechanical Engineers (ASME) RA–Sb–2005, "Addenda to ASME RA–S–2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," (Reference 6), NEI 00–02, "PRA Peer Review Process Guidelines," (Reference 7) and NEI 05–04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard" (Reference 8). The licensee has performed an assessment of the PRA models used to support the SFCP against the requirements of RG 1.200 to assure that the PRA models are 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 ASME RA–Sb–2005 is applied as the standard, and any identified deficiencies to those requirements are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including by the use of sensitivity studies where appropriate.

The NRC staff reviewed the licensee's assessment of the OCNGS PRA and the remaining open deficiencies that do not conform to Capability Category II of the ASME PRA standard (Table 2-1 of Attachment 2 of the license amendment request). The NRC staff's assessment of these open "gaps," to assure that they may be addressed and dispositioned for each surveillance frequency evaluation per the NEI 04-10 methodology, is provided below.

Gap #1: The PRA applies a 24-hour mission time for some longer term loss of decay heat removal sequences where the time to core damage exceeds 24 hours. The licensee stated that the OCNGS evaluation of these sequences is reasonable given that repair and recovery of failed equipment is not considered for these sequences, and this would offset the increased failure probabilities which would conservatively result from extending the 24-hour mission time. The NRC staff agrees that this potential non-conservatism will not significantly impact results for this application; therefore, this deficiency can be addressed per the methodology of NEI 04-10.

Gap #2: Some system notebooks do not include plant staff experience or procedure references. The licensee determined this to be a documentation issue which is being resolved. The NRC staff concurs with this assessment.

Gaps #3, #4: The failure modes included in the model may be incomplete, and the criteria for exclusion of failure modes based on low probability are not documented. The licensee identified that active failure modes which are impacted by surveillance frequency changes are included in the model, and that documentation enhancements are needed to close these gaps; therefore, the NRC staff concurs that these deficiencies can be addressed per the methodology of NEI 04-10.

Gap #5: The basic event nomenclature is inconsistent with other Exelon PRA models. This documentation issue has no impact on the technical adequacy of the PRA model. The NRC staff concurs with this assessment.

Gap #6: The system notebooks do not include some documentation requirements of supporting requirement SY-C2. The licensee stated that this is a documentation issue which has no impact on the technical adequacy of the PRA model. The NRC staff concurs with this assessment.

Gap #7: Supporting documentation relating the plant procedures to the pre-initiator actions is required by supporting requirement HR-B1. The licensee stated that this is only a documentation issue that has no impact on the technical adequacy of the PRA model. The NRC staff concurs with this assessment.

Gap #8: The quality of the written procedures and the human-machine interface for both pre- and post-initiator actions had not been assessed per supporting requirement HR-D3. Subsequent to the initial submittal of this license amendment request, and in response to a request for additional information (RAI), the licensee assessed the OCNGS procedures, administrative controls, and the human-machine interface as they relate to both pre- and post-initiator actions. Based on its internal assessment, the licensee found that the quality exceeds the assumptions made in the human reliability analyses for the PRA model and no changes were required. Therefore, the NRC staff finds that this gap has been properly addressed to assure conformance to the standard requirements.

Gaps #9, #10, #11, and #12: Data to support plant-specific demands, standby times, and component failures are based in part on estimations. The licensee identified in response to an RAI that the number of component failures is based on actual data, but that the number of demands or standby time is estimated based on scheduled testing. This may lead to slight conservatism in the resulting failure rates due to unscheduled testing, and the NEI 04-10 methodology addresses this via sensitivity studies on failure probabilities. The NRC staff concurs that the method used to evaluate plant-specific failure rates is acceptably conservative; therefore, this deficiency can be addressed per the methodology of NEI 04-10.

Gap #13: Maintenance unavailability data uncertainty bounds were not supplemented by plant staff interviews. The licensee stated that interviews were not used to generate input for parametric uncertainties, and actual data was used. Since the supporting requirement DA-C13 only requires interviews when the maintenance times are not based on plant data, the NRC staff finds the disposition of this gap acceptable.

Gap #14: The internal flooding analysis flood areas were not identified consistent with supporting requirement IF-A1a. The licensee described its method for determining separate flood areas in the plant based on timing and consequences of postulated internal flood initiators. The focused scope peer review of internal flooding, conducted in August 2008, specifically reviewed this supporting requirement for OCNGS and identified documentation enhancements to resolve the deficiency. Based on the licensee's description of its compartment boundaries and the peer review findings, the NRC staff concurs that the flood areas are acceptably defined to support the risk analysis.

Gap #15: The internal flood analysis does not include plant-specific data or events. The licensee stated that no significant events have occurred at OCNGS, and so inclusion of plant-specific operating experience is expected to have a negligible impact. This deficiency can, therefore, be addressed per the methodology of NEI 04-10.

Gaps #16, #17, #18, and #19: These gaps address deficiencies in the documentation of the quantification of the PRA and the LERF analysis, and therefore, do not directly impact the technical adequacy of the PRA model.

Based on the licensee's assessment using the applicable PRA standard and RG 1.200, the NRC staff finds that the level of PRA quality, combined with the proposed 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.

3.1.4.2 Scope of the PRA

The licensee is required to evaluate each proposed change to a relocated surveillance frequency using the guidance contained in NEI 04-10 to determine its potential impact on risk, due to impacts from internal events, fires, seismic, other external events, and from shutdown conditions. Consideration is made of both CDF and LERF metrics. In cases where a PRA of sufficient scope or where 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 has developed a fire PRA model, which is an interim implementation of the methodology of NUREG/CR-6850 because not all tasks identified in this document are completely addressed. For surveillance interval evaluations, the licensee intends to employ a limited use of the fire PRA model to obtain quantitative insights when needed to supplement a qualitative or bounding assessment.

The individual plant examination of external events (IPEEE) seismic PRA will be used to provide seismic insights. Other external hazards were screened during the IPEEE assessment, and therefore, will be qualitatively assessed for this application.

Based on the above evaluation, the NRC staff finds that licensee's evaluation methodology is sufficient to ensure the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation. Therefore, the scope of the PRA is consistent with regulatory position 2.3.2 of RG 1.177.

3.1.4.3 PRA Modeling

The licensee will determine 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 common cause failure 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. The licensee will perform quantitative evaluations of the impact of selected testing strategy (i.e., staggered testing or sequential testing) consistent with the guidance of NUREG/CR–6141 and NUREG/CR–5497, as discussed in NEI 04–10.

Thus, through the application of NEI 04–10 the OCNGS 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.

3.1.4.4 Assumptions for Time Related Failure Contributions

The failure probabilities of SSCs modeled in the OCNGS PRA include a standby time-related contribution and a cyclic demand-related contribution. NEI 04-10 criteria adjust the time-related failure contribution of SSCs affected by the proposed change to surveillance frequency. This is consistent with RG 1.177 Section 2.3.3 which permits separation of the failure rate contributions into demand and standby for evaluation of SRs. If the available data do not support distinguishing between the time-related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, and will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The 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 beneficial risk impacts of reduced surveillance frequency, including reduced downtime, lesser potential for restoration errors, reduction of potential for test caused transients, and reduced test-caused wear of equipment, are identified qualitatively, but are conservatively not required to be quantitatively assessed. Thus, through the application of NEI 04-10, the licensee has employed reasonable assumptions with regard to extensions of surveillance test intervals, and is consistent with Regulatory Position 2.3.4 of RG 1.177.

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04-10 requires sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact to the frequency of initiating events, and of any identified deviations from capability category II of ASME PRA Standard (ASME RA–Sb–2005) (Reference 4). 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. Required monitoring and feedback of SSC performance once the revised surveillance frequencies are implemented will also be performed. Thus, through the application of NEI 04-10, 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.

3.1.4.6 Acceptance Guidelines

The licensee will 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 the guidance contained in NRC approved NEI 04-10 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 are consistent with the limits of RG 1.174 for very small changes in risk. Where the RG 1.174 limits are not met, the process either considers revised surveillance frequencies which are consistent with RG 1.174 or the process terminates without permitting the proposed changes. Where quantitative results are unavailable to permit comparison to 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 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 below 1E–5 per year for change to CDF, and below 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 are consistent with the limits of RG 1.174 for acceptable changes in risk, as referenced by RG 1.177 for changes to surveillance frequencies. The NRC staff interprets this assessment of cumulative risk as a requirement to calculate the change in risk from a baseline model utilizing failure probabilities based on the surveillance frequencies prior to implementation of the SFCP, compared to a revised model with failure probabilities based on changed surveillance frequencies. The NRC staff further notes that Exelon included a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with small risk increases (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 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 compared to numerical acceptance guidelines. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. The licensee's application of NEI 04-10 provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177. Therefore, the proposed Exelon methodology satisfies the fourth key safety principle of RG 1.177 by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement.

3.1.5 <u>The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies</u>

The licensee's proposed amendment requires application of NEI 04-10 in the SFCP. NEI 04-10 requires performance monitoring of SSCs whose surveillance frequency has 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 degradation of SSC performance, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions which may apply as part of the maintenance rule requirements. The performance monitoring and feedback specified in NEI 04-10 is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177. Thus, the fifth key safety principle of RG 1.177 is satisfied.

3.2 <u>Addition of Surveillance Frequency Control Program to Administrative Controls</u>

The licensee has included the SFCP and specific requirements into the Administrative Controls, TS Section 6.24, Surveillance Frequency Control Program, as follows:

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 the 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 Method for Control of Surveillance Frequencies," Revision 1.*

c. *The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.*

The NRC staff finds that the proposed addition to the Administrative Controls section of the TSs adequately identifies scope of the SFCP and defines the methodology to be used in revision of SR frequencies.

3.3 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a licensee-controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the administrative controls of TS. The SFCP and TS Section 6.24 references NEI 04-10, 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 TS to a licensee-controlled document, provided those frequencies are changed in accordance with NEI 04-10 which is specified in the Administrative Controls of the TS.

The proposed licensee adoption the risk-informed methodology of NEI 04-10 as referenced in the Administrative Controls Section of the TS, satisfies the key principles of risk-informed decision making applied to changes to TS 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.

10 CFR 50.36(c)(3) states "Technical specifications will include items in the following categories: Surveillance Requirements. 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 NRC staff finds that with the proposed relocation of surveillance frequencies to an owner-controlled document and administratively controlled in accordance with the TS SFCP, Exelon continues to meet the regulatory requirement of 10 CFR 50.36, and specifically, 10 CFR 50.36(c)(3), SRs.

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

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

3.4 TS Bases

Because the Oyster Creek TS Bases are integrated into the TS vice being contained in a separate section, some of the issued pages contain bases revisions associated with the proposed change. The revised wording in the bases is included only for ease of implementation and does not imply NRC staff review or approval of their content.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes 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 that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (74 FR 68869). Accordingly, the 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 the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b," March 18, 2009 (ADAMS Accession No.: ML090850642).

2. NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession No.: ML071360456).

3. Regulatory Guide 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' August 1998 (ADAMS Accession No.: ML003740176).

4. Regulatory Guide 1.200, ''An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,'' Revision 1, January 2007 (ADAMS Accession No.: ML070240001).

5. Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant- Specific Changes to the Licensing Basis,'' NRC, Revision 1, November 2002 (ADAMS Accession No.: ML023240437).

6. ASME PRA Standard ASME RA-Sb-2005, "Addenda to ASME RA–S–2002, ''Standard for Probabilistic Risk Assessment for Nuclear Power Plant Application.''

7. NEI 00-02, Revision 1 ''Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, Revision 1, May 2006 (ADAMS Accession No.: ML061510621).

8. NEI 05-04, ''Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard'', Revision 0, August 2006.

9. P. B. Cowan to U. S. NRC, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," October 30, 2009. (ADAMS Accession No.: ML093060126).

10. P. B. Cowan to U. S. NRC, "Response to Request for Additional Information, License Amendment Request Regarding Relocation of Selected Technical Specification Surveillance Frequencies to a Licensee Controlled Program," April 16, 2010. (ADAMS Accession No.: ML101060560).

11. P. B. Cowan to U. S. NRC, "Response to draft Request for Additional Information, License Amendment Request Regarding Relocation of Selected Technical Specification Surveillance Frequencies to a Licensee Controlled Document," August 31, 2010. (ADAMS Accession No. ML102430467).

12. H. K. Nieh, to B. Bradley, Nuclear Energy Institute, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," September 19, 2007. (ADAMS Accession No. ML072570267).

Principal Contributor: A. Howe

Date: September 27, 2010

September 27, 2010

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: RELOCATION OF SURVEILLANCE REQUIREMENT FREQUENCIES TO A LICENSEE CONTROLLED DOCUMENT BASED ON TECHNICAL SPECIFICATION TASK FORCE-425, REVISION 3 (TAC NO. ME2494)

Dear Mr. Pacilio:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 276 to Renewed Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek), in response to your application dated October 30, 2009, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093060126), as supplemented by letter dated April 16, 2010, and August 31, 2010 (ADAMS Accession Nos. ML101060560 and ML102430467, respectively). The amendment revises the Oyster Creek Technical Specifications to relocate a number of Surveillance Requirement frequencies to a licensee controlled document.

A copy of our Safety Evaluation is enclosed and a Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/ra/
G. Edward Miller, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures:
1. Amendment No. 276 to Renewed DPR-16
2. Safety Evaluation

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GHill (2)

Accession No: ML101930172 *Via SE Input

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NAME	GEMiller	ABaxter	DHarrison*	RElliott	A. Jones	HChernoff
DATE	7/23/10	9/8/10	5/2/10	8/5/10	8/16/10	9/27/10

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