NRR-DMPSPEm Resource

From: Lamb, John
Sent: Friday, August 10, 2018 12:34 PM
To: Schwartz, Paul; Gubbi, Veena
Subject: For Your Comments - State of New Jersey - Oyster Creek License Amendment Request regarding Permanently Defueled Technical Specifications
Attachments: LTR - Oyster Creek PDTS LAR - 11-16-2017.pdf; LTR - Oyster Creek PDTS LAR - 3-29-2018.pdf

Importance: High

Mr. Schwartz and Ms. Gubbi:

The U.S. Nuclear Regulatory Commission plans to issue an Amendment to Renewed Facility Operating License (RFOL) No. DPR-16 for Oyster Creek Nuclear Generating Station (Oyster Creek) approximately on September 28, 2018, in response to the Exelon application dated November 16, 2017 (ML17320A411), as supplemented by letter dated March 29, 2018 (ML18088A317).

The proposed amendment would revise the RFOL and the associated Technical Specifications to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

If the State of New Jersey has any comments regarding this license amendment request, please e-mail me your comments by the close of business on September 14, 2018.

Thanks.
John

Hearing Identifier: NRR_DMPS
Email Number: 513

Mail Envelope Properties (John.Lamb@nrc.gov20180810123400)

Subject: For Your Comments - State of New Jersey - Oyster Creek License Amendment Request regarding Permanently Defueled Technical Specifications
Sent Date: 8/10/2018 12:34:10 PM
Received Date: 8/10/2018 12:34:00 PM
From: Lamb, John

Created By: John.Lamb@nrc.gov

Recipients:
"Schwartz, Paul" <Paul.Schwartz@dep.nj.gov>
Tracking Status: None
"Gubbi, Veena" <Veena.Gubbi@dep.nj.gov>
Tracking Status: None

Post Office:

Files	**Size**	**Date & Time**
MESSAGE	828	8/10/2018 12:34:00 PM
LTR - Oyster Creek PDTS LAR - 11-16-2017.pdf	4690379	
LTR - Oyster Creek PDTS LAR - 3-29-2018.pdf	2955753	

Options
Priority: High
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Exelon Generation®

Michael P. Gallagher
Exelon Nuclear
Vice President
License Renewal and Decommissioning

200 Exelon Way
Kennett Square, PA 19348

610 765 5958 Office
610 765 5658 Fax
www.exeloncorp.com
michaelp.gallagher@exeloncorp.com

10 CFR 50.90

RA-17-072

November 16, 2017

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
NRC Docket No. 50-219 and 72-15

Subject: License Amendment Request – Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition

1. Letter from Keith R. Jury, Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission - "Permanent Cessation of Operations at Oyster Creek Nuclear Generating Station," dated January 7, 2011 (ML110070507)
2. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC) – "Oyster Creek Nuclear Generating Station - Issuance of Amendment Regarding Changes to The Administrative Controls Section Of The Technical Specifications (CAC NO. MF8108)," dated March 7, 2017 (ML16235A413)
3. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station – Issuance of Amendment RE: Deletion of Facility Operating License Conditions Related to Decommissioning Trust Provisions (CAC No. MF9293)," dated June 23, 2017 (ML17067A042)
4. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station; Clinton Power Station, Unit No. 1; and Quad Cities Nuclear Power Station, Units 1 And 2 – Approval of Certified Fuel Handler Training and Retraining Program (CAC NOS. MF8109, MF8138, MF8139, AND MF8140)," dated September 6, 2016 (ML16222A787)
5. Letter from David P. Helker (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "License Amendment Request Regarding Revision to Cyber Security Plan Milestone 8 Completion Date," dated April 10, 2017 (ML17100A844).

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (Exelon) requests amendments to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. DPR-16 for Oyster Creek Nuclear Generating Station (OCNGS). The proposed amendment would revise the RFOL and the associated TS to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

By letter dated January 7, 2011 (Reference 1), Exelon provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) of Exelon's contingent determination to permanently cease operations at OCNGS no later than December 31, 2019. Once the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC pursuant to 10 CFR 50.82(a)(1)(i) and (ii), NRC regulations stipulated in 10 CFR 50.82(a)(2) will no longer authorize operation of the reactor or placement of fuel into the reactor vessel under the 10 CFR 50 license. In support of this condition, the OCNGS RFOL and associated TS are being proposed for revision to reflect the planned permanently shutdown and defueled condition in accordance with 10 CFR 50.36(c)(6).

The bases for the proposed amendment is that certain license conditions and TS requirements may be revised or removed to reflect the permanently defueled condition. In general, the changes propose the elimination of those TS applicable in operating modes or modes where fuel is placed in the reactor vessel. Changes to other TS limiting conditions for operation, definitions, surveillance requirements, administrative controls, as well as several license conditions are also proposed. The proposed amendment would modify the 10 CFR Part 50 License and the TS to make those changes.

The NRC has already approved previous requests for changes to the organization, staffing, and training requirements contained in Section 6, "Administrative Controls" of the OCNGS TS (Reference 2), the License Conditions for maintaining the Decommissioning Trust Fund (Reference 3), and the Certified Fuel Handler Training and Retraining Program (Reference 4).

In Reference 5, Exelon proposed a change to the OCNGS Cyber Security Plan Milestone 8 full implementation date. This date change is reflected in the Security Plan license condition in the OL. The four referenced licensing actions complement and support this proposed license amendment.

Exelon has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92.

The proposed changes have been reviewed and approved by the station's Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Attachment 1 to this letter provides a detailed description and evaluation of the proposed change. Attachment 2 contains a markup of the current OL and TS pages, including Bases (TS sections that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 2).

Exelon requests review and approval of this proposed amendment by November 16, 2018, and a 60-day implementation period following the effective date of the amendment. Exelon requests

that the approved amendment become effective following submittal of the required 10 CFR 50.82(a)(1)(ii) certification that OCNGS has been permanently defueled. Once effective, implementation will occur within the 60 days, as noted, but will not exceed March 29, 2020.

There are no regulatory commitments contained within this submittal.

In accordance with 10 CFR 50.91 "Notice for public comment; State consultation" paragraph (b), Exelon is notifying the State of New Jersey of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions concerning this submittal, please contact Paul Bonnett at (610) 765-5264.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of November 2017.

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President, License Renewal & Decommissioning
Exelon Generation Company, LLC

Attachments: 1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specifications Pages

cc: w/Attachments

Regional Administrator - NRC Region I
NRC Senior Resident Inspector - Oyster Creek Nuclear Generating Station
NRC Project Manager, NRR - Oyster Creek Nuclear Generating Station
Director, Bureau of Nuclear Engineering - New Jersey Department of Environmental Protection
Mayor of Lacey Township, Forked River, NJ

Attachment 1

License Amendment Request

Oyster Creek Nuclear Generation Station

Docket Nos. 50-219 and 72-15

EVALUATION OF PROPOSED CHANGES

Subject: Proposed Changes to Renewed Facility Operating License and Appendix A, Technical Specifications

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon), proposes an amendment to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. DPR-16 for Oyster Creek Nuclear Generation Station (OCNGS). The proposed amendment would revise the RFOL and the associated TS to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

On December 9, 2010, Exelon and the New Jersey Department of Environmental Protection (NJDEP) executed an Administrative Consent Order (ACO). Under the terms of this Order, Exelon agreed to permanently cease operations at OCNGS no later than December 31, 2019. By letter dated January 7, 2011 (Reference 1), Exelon informed the U.S. Nuclear Regulatory Commission (NRC) of Exelon's contingent determination to permanently cease operations at OCNGS no later than December 31, 2019.

Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii), pursuant to 10 CFR 50.82(a)(2), the 10 CFR 50 license for OCNGS will no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel. As a result, OCNGS will be authorized to only possess special nuclear material. In support of this condition, the OCNGS RFOL and associated TS are being proposed for revision to reflect the planned permanent shutdown and defueled condition pursuant to 10 CFR 50.36(c)(6) "*Decommissioning*."

The proposed License Amendment Request (LAR) revise and remove certain requirements contained within the RFOL and TS, and remove the requirements that would no longer be applicable after the OCNGS reactor has been permanently defueled. The proposed changes to the RFOL and TS are in accordance with 10 CFR 50.36(c)(1) through (c)(5). The proposed changes include revised formatting, numbering, and wording where appropriate, to condense the number of pages in the TS without affecting the technical content. The TS Table of Contents is also accordingly revised.

The current OCNGS TS have been customized over the years to meet the specific needs of the unit. These TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including safe storage and management of irradiated fuel. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a permanently defueled plant.

In the development of the proposed PDTS changes, Exelon reviewed the PDTS requirements from other plants that have permanently shutdown, primarily Vermont Yankee (Reference 9), Kewaunee (Reference 10), San Onofre Nuclear Generating Station (Reference 11), and Crystal River Nuclear Plant, Unit 3 (Reference 12). Exelon also evaluated the applicable guidance in NUREG-1433, *"Standard Technical Specifications General Electric BWR/4 Plants"* (Reference 8) and Draft NUREG-1625, *"Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants"* (Reference 13).

This LAR provides a discussion and description of the proposed RFOL and TS changes, a technical evaluation of the proposed RFOL and TS changes, and information supporting a finding of No Significant Hazards Consideration (NSHC).

Pending Licensing Actions under NRC Review

By letter dated September 6, 2016 (Reference 2), the NRC approved the Certified Fuel Handler (CFH) Training and Retraining Program for OCNGS. By letter dated March 7, 2017 (Reference 3), the NRC issued License Amendment Number 290 for proposed changes to the organization, staffing, and training requirements contained in OCNGS TS Section 6, Administrative Controls, which is incorporated into the LAR. The CFH program and the TS amendment will become effective and be implemented once the OCNGS reactor has been completely defueled and the certification of permanent removal of fuel from the reactor vessel has been docketed pursuant to 10 CFR 50.82(a)(1)(ii).

By letter dated June 23, 2017, the NRC issued License Amendment Number 291 to the RFOL that deleted certain license conditions which imposed specific requirements on the Decommissioning Trust Fund (DTF) agreement (Reference 4). Based on this amendment, the provision of 10 CFR 50.75(h) that specify the regulatory requirements for DTF will apply to OCNGS.

There are currently two other pending license amendment requests involving proposed changes to TS currently docketed for OCNGS. One proposed amendment is a request to change the Implementation Milestone Regulatory Commitment for the Exelon Cyber Security Plan (CSP) Milestone 8 full implementation date as set forth in the Exelon CSP Implementation Schedule (Reference 5). The second proposed license amendment is to revise RFOL Section 2.C, License Condition (5) to adopt "BWR Vessel and Internals Project, BWR Core Spray Internals inspection and Flaw Evaluation Guidelines," BWRVIP-18, Revision 2-A (Reference 19). The proposed change is to be utilized in the final refueling outage in the Fall of 2018.

The above mentioned licensing actions complement and support this proposed license amendment.

2.0 DETAILED DESCRIPTION AND BASIS FOR THE CHANGES

The proposed amendment would modify the OCNGS RFOL and transform the operating OCNGS TS into the OCNGS PDTS to comport with a permanently defueled condition.

General Analysis Applicable to Proposed Change

Chapter 15 of the OCNGS Updated Final Safety Analysis Report (UFSAR) describes the design basis accidents (DBA) and transient scenarios applicable to OCNGS during power operations. During normal power operations, the forced inlet flow of water through the reactor coolant system (RCS) removes the heat from the reactor by generating steam. The steam system, operating at high temperatures and pressures, transfers this heat to the turbine generator. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the reactor coolant system. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core.

With the termination of reactor operations at OCNGS and the permanent removal of fuel from the reactor pressure vessel (RPV) as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the DBA scenarios postulated in the UFSAR are no longer be possible. During decommissioning, the irradiated fuel will be stored in the Spent Fuel Pool (SFP) and the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped off site

in accordance with the schedules to be provided in the Post Shutdown Decommissioning Activities Report (PSDAR) and the Spent Fuel Management Plan. The RCS, steam system, and turbine generator are no longer in operation and have no function related to the safe storage and management of the spent nuclear fuel.

Chapter 15 of the OCNGS UFSAR describes the safety analysis aspects of the plant that were evaluated to demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed the guidelines of 10 CFR 50.67, *"Accident source term."* Two basic groups of events pertinent to safety, abnormal operational transients, and DBAs were investigated separately. The analyses of the abnormal operational transients evaluate the ability of the plant protection features to ensure that, during these transients, no fuel damage occurs and the RCS pressure limit is not exceeded. For DBAs, the analyses evaluate the ability of the plant, with its various "containment" barriers and engineered safeguards, to withstand the effects of these postulated accidents and to minimize offsite radiological consequences.

Safety analyses are evaluated against regulatory acceptance criteria and are integral of the plant's design and licensing basis. The safety analyses demonstrate the integrity of the fission product barriers, the capability to shutdown the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents and transients. Systems, Structures, and Components (SSCs) that perform design basis functions are credited in the safety analyses for the purpose of mitigating the transient or accident.

A list of the OCNGS UFSAR Chapter 15 DBAs and whether the accident applies to a permanently defueled condition is provided in Table 2.1.

TABLE 2.1 – OCNGS Design Basis Accidents

Postulated Accident or Transient	Defueled Applicability
15.1 Increase in Heat Removal by the Secondary System	
15.1.1 Decrease in Feedwater Temperature	Not Applicable
15.1.2 Feedwater Controller Failure	Not Applicable
15.1.3 Increase in Steam Flow	Not Applicable
15.1.4 Inadvertent Opening of a Steam Generator Relief/Safety Valve	Not Applicable
15.1.5 Steam System Piping Failures Outside Containment	Not Applicable
15.2 Decrease in Heat Removal by the Secondary System	
15.2.1 Steam Pressure Regulator Failure	Not Applicable
15.2.2 Loss of External Load	Not Applicable
15.2.3 Turbine Trip (Stop Valve Closure)	Not Applicable
15.2.4 Inadvertent Closure of Main Steam Isolation Valves	Not Applicable
15.2.5 Loss of Condenser Vacuum	Not Applicable
15.2.6 Loss of All AC Power/Loss of Auxiliary Power	Not Applicable
15.2.7 Loss of Normal Feedwater Flow	Not Applicable
15.2.8 Feedwater Pipe Break	Not Applicable
15.2.9 Loss of Stator Cooling	Not Applicable
15.3 Decrease in Reactor Coolant System Flow Rate	
15.3.1 Single and Multiple Recirculation Pump Trips	Not Applicable
15.3.2 Recirculation Flow Controller Malfunctions (Flow Decrease)	Not Applicable

	15.3.3 Recirculation Pump Stall (Shaft Seizure)	Not Applicable
	15.3.4 Recirculation Pump Shaft Break	Not Applicable
15.4	Reactivity and Power Distribution Anomalies	
	15.4.1 Uncontrolled Control Rod Withdrawal at Startup/Low Power	Not Applicable
	15.4.2 Uncontrolled Control Rod Withdrawal at Power	Not Applicable
	15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)	Not Applicable
	15.4.4 Startup of an Inactive Loop at an Incorrect Temperature	Not Applicable
	15.4.5 Flow Controller Malfunction (Increase in Core Flow)	Not Applicable
	15.4.6 Chemical and Volume Control System Malfunctions	Not Applicable
	15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Not Applicable
	15.4.8 Spectrum of Rod Ejection Accidents	Not Applicable
	15.4.9 Control Rod Drop Analysis	Not Applicable
15.5	Increase in Reactor Coolant Inventory	
	15.5.1 Inadvertent Operation of ECCS During Power Operation	Not Applicable
	15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Not Applicable
	15.5.3 Other BWR Transients (including Items 2.1 through 2.6, and Item 1.2)	Not Applicable
15.6	Decrease In Reactor Coolant Inventory	
	15.6.1 Inadvertent Opening of a Safety or Relief Valve	Not Applicable
	15.6.2 Break in an Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrates Containment	Not Applicable
	15.6.3 Steam Generator Tube Failure	Not Applicable
	15.6.4 Steam Line Failures Outside of Containment	Not Applicable
	15.6.5 Loss-of-Coolant Accidents Resulting from the Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Not Applicable
	15.6.6 A Number of BWR Transients	Not Applicable
15.7	Radioactive Release from a Subsystem or Component	
	15.7.1 Waste Gas System Rupture Accident	Not Applicable
	15.7.2 Radioactive Liquid Waste System Leak or Failure	Applicable
	15.7.3 Postulated Radioactive Releases Due to Liquid Tank Failure	Applicable
	15.7.4 Design Basis Fuel Handling Accidents in the Containment	Applicable
	15.7.5 Spent Fuel Cask Drop Accident	Not Credible
15.8	Anticipated Transients Without Scram (ATWS)	
	15.8.1 Alternate Rod Insertion (ARI)	Not Applicable
	15.8.2 Standby Liquid Control System	Not Applicable
	15.8.3 Recirculation Pump Trip (RPT)	Not Applicable
15.9	Station Blackout	Not Applicable

The analyzed accidents that remains applicable to OCNGS in the permanently shut down and defueled condition is a Fuel Handling Accident (FHA) in the SFP (a dropped fuel assembly onto the top of the core will no longer be applicable) and the Postulated Radioactive Tank Failure and Release of Radioactive Liquid Waste while radioactive liquids are still present. As discussed in Section 15.7.2 and 15.7.3 of the UFSAR, the analysis for the Liquid Release, assuming failure of all liquid radwaste equipment, would result in a computed dose due to noble gases not to exceed 500 mrem at the site boundary. These analyses remain valid and therefore, these accidents will not be further addressed in this submittal.

10 CFR 50.36, *"Technical specifications,"* promulgates the regulatory requirements related to the content of Technical Specifications. As detailed in subsequent sections of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS (see Section 3.1). In a permanently defueled condition, the scope of equipment and parameters that must be included in the OCNGS PDTS is limited to those needed to address the remaining applicable DBA (i.e., the postulated FHA) so that the consequences of the accident are maintained within acceptable limits.

Fuel Handling Accident Analysis for the Permanently Defueled Condition

The FHA in the SFP is the remaining Chapter 15 accident with a radiological consequence to the health and safety of the public and control room (CR). A calculation (C-1302-226-E310-460, "EAB, LPZ, and CR Dose Due to Fuel Handling Accident (FHA) - Post Cessation of Power Operations" (Reference 6)) was performed to assess the dose consequences of a postulated FHA after cessation of power operations. The calculation demonstrates that radiological doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the CR are within allowable limits of 10 CFR 50.67 without crediting secondary containment operability, standby gas treatment system, or CR high efficiency air filtration after a 60-day fuel decay period following permanent reactor shutdown. This calculation was performed using the methodology currently described in the OCNGS UFSAR, which is based on the Alternative Source Term (AST) defined in Regulatory Guide 1.183, *"Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"* (Reference 14).

Regulatory Guide 1.183 provides guidance for the implementation of the AST methodology for DBAs. 10 CFR 50.67 requires licensees seeking to use the AST methodology to apply for a license amendment and include an evaluation of the consequences of the affected design-basis accidents. This license amendment request addresses these requirements by proposing a limited scope application of the AST discussed in Regulatory Guide 1.183 for evaluating the radiological consequences of an FHA. As part of the implementation of the AST methodology, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) is applied (see Section 3.1).

The accident is assumed to occur during handling of a spent fuel assembly in the SFP. The spent fuel assembly is assumed to drop and strike assemblies stored in the SFP. There is slightly over 23 feet of water above the top of active fuel for bundles within the storage racks; however, this does not ensure the dropped bundle will have 23 feet of water coverage above it. As described in the FHA in the RPV, the amount of fuel that fails is assumed to correspond to a dropped fuel assembly from 30 feet onto the reactor core during refueling. This amount of fuel damage is significantly higher than a drop in the SFP, which corresponds to a significantly shorter fall (approximately 4 feet). The amount of fuel failures assumed in the FHA in the RPV was conservatively used in the FHA in the SFP. Due to the reduced drop height, the amount of fuel failures would be about 50% less than the FHA in the RPV. The decontamination factor (DF) associated with instantaneous release of noble gases and iodine from the fuel would be maintained because less fuel damage would occur due to the shorter drop compensating for having slightly less than 23 feet of water coverage above the dropped bundle. The remaining

airborne activity is assumed to release to the environment over a 2-hour period as prescribed in Regulatory Guide 1.183. The release is assumed to occur at ground-level near the location of Main Steam Isolation Valves (MSIVs) in the turbine building. This release point was selected because it does not credit any safety-system and it conservatively bounds the atmospheric dispersion from the fuel building to the EAB, LPZ, and any CR intake and was previously evaluated to address the ground-level MSIV leakage during the AST Loss of Coolant Accident (LOCA) analysis.

The CR dose model consists of a finite volume and assumes amounts of infiltration and exfiltration of the activity released to the environment. There is no credit for filtration in the CR ventilation model. To address downgrading safety-related features of the CR ventilation system, a sensitivity study on unfiltered intake flow rate and CR isolation time was performed. This study determined that the maximum CR dose consequences occur after the CR is isolated (10 cfm inflow or outflow assumed to account for normal operator ingress and egress to the CR) at 5 minutes post-accident. This assumption bounds any scenario where the CR ventilation system fails and traps activity inside.

The dose consequences were calculated using the RADTRAD computer software. It was determined that 60 days of decay was adequate to meet the dose consequence limits. The results are summarized below:

Receptor	Dose (rem TEDE)	Limit
Control Room	2.2349E+00	5
Exclusion Area Boundary	1.5170E-02	6.3
Low Population Zone	1.4524E-03	6.3

The resulting dose consequences are within the limits of 10 CFR 50.67 and Regulatory Guide 1.183. The analysis does not credit any safety systems and assumes very conservative atmospheric dispersion. There are no operator actions assumed in this analysis and there is no reliance on any radiation monitoring or heating, ventilation, and air conditioning (HVAC) system actuations. The analysis conservatively assumes a large unfiltered in-leakage into the CR and accounts for unplanned stoppage of the ventilation flow in and out of the CR.

The analysis demonstrates that radiological doses at the EAB, LPZ, and in the CR from a FHA after 60 days following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system. No equipment is required to mitigate the effects of this event beyond the administrative controls described in Reference 6.

Monitoring of Neutron Absorbing Material – Spent Fuel Pool

OCNGS has a Neutron Absorber Monitoring Program for monitoring and maintaining the neutron absorbing materials in the SFP during decommissioning. OCNGS is currently licensed to store 3035 fuel bundles in 10 Boraflex high density storage racks and four Boral high density racks. The design of the spent fuel storage racks provides for a subcritical multiplication factor K_{eff} of less than or equal to 0.95 for normal and abnormal storage conditions.

The OCNGS RFOL includes a license condition (2.C.(16)) that requires completion of Exelon's commitments that are described in Appendix A to NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Nuclear Generating Station"* (Reference 7). Commitment 15 (NUREG-1875, Appendix A) credits the existing program for Boraflex Rack Monitoring Program at OCNGS. NUREG-1875, Section 3.0.3.2.13, discusses the neutron-absorbing material (NAM) monitoring program for Boraflex spent fuel racks at OCNGS. The NRC reviewed and approved the

Oyster Creek Boraflex Rack Management Program and the UFSAR description of the program as part of the license renewal review. Further, Exelon provided the NRC with additional information in response to NRC Generic Letter (GL) 2016-01, "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools" (Reference 15).

The Boraflex Rack Management Program provides for aging management of the Boraflex neutron poison material. The program consists of monitoring the condition of Boraflex by routinely sampling fuel pool silica levels and periodically performing in-situ measurement of boron-10 areal density using the BADGER device. RACKLIFE is used as an administrative tool to trend the condition of the Boraflex. As discussed in Exelon's response to the GL, OCNGS made one change to the Boraflex Rack Monitoring Program documented in the License Renewal Application and in NUREG-1875. The station stopped using Boraflex coupon surveillances as a part of the program. This was done because the Boraflex coupons became non-representative of the in-service material according to the BADGER testing results, as experience by other Boraflex users in the industry. This change does not impact the effectiveness of the Boraflex Rack Management Program. No further change has been made to the Boraflex Rack Management Program as documented in the License Renewal Application.

OCNGS utilizes periodic coupon measurements to monitor the ability of the Boral to perform its safety function. Coupon material monitoring methods rely on the strong correlation between aging/degradation impacts on a set of surrogate material pieces (coupons) from the same manufacturing process as the as-installed material. The station maintains the ability to detect aging/degradation mechanisms that the in-service NAM experiences through monitoring the coupon material characteristics. Coupon trees are maintained in SFP locations that have conditions projected to be the most challenging to the materials (e.g., high gamma dose, high neutron dose, high temperature) to ensure early detection of aging/degradation mechanisms that are driven by environmental factors.

OCNGS utilizes a Boral coupon surveillance program to confirm that the material is performing its safety function. The current surveillance program is documented in site procedures. This surveillance plan has been structured to meet the guidance equivalent to Section XI.M40 of the Generic Aging Lessons Learned (GALL) Report, (NUREG-1801, Revision 2) (Reference 16). The plan was last updated based on the plant's response to NRC Information Notice 2009-26 (Reference 17). OCNGS has implemented the requirement for inspecting a minimum of one Boral coupon every 10 years. This periodicity is sufficient to demonstrate that the Boral material can perform the design function of keeping the spent fuel arrangement at 5 percent margin to criticality. This is based on the vast industry experience with Boral that has not shown any currently known mechanism that leads to the loss of the boron from the Boral material. This sampling frequency is in agreement with the guidance in Section XI.M40 of the GALL Report. Exelon also monitors industry experience with the Boral NAM through operating experience reviews and through industry group participation (e.g., NEI, EPRI).

Exelon is proposing to change TS Section 3.1, "Protective Instrumentation" to be TS Section 3/4.1 "Spent Fuel Storage" with a new proposed TS LCO: TS 3.1, "Spent Fuel Pool Water Level."

Detailed Discussion

The following tables identify each RFOL and TS section that is being changed, the proposed change, and the basis for each change. Changes to the RFOL are addressed first, followed by the TS. Proposed revisions are shown in ***Bold-Italics*** and deletions are shown using ~~strikethrough~~. Proposed changes to the TS Bases addressing the proposed changes to the relevant TS are provided for information in Attachment 2. Upon approval of this amendment,

changes to the Bases will be incorporated in accordance with TS 6.21 TS Bases Control Program, which is retained in its entirety without change.

Attachment 2 provides the marked-up version of the OCNGS RFOL and TS to establish the changes. The TSs that are deleted in their entirety are identified as such below, but the associated deleted pages are not included in Attachment 2. Additionally, the proposed changes to the TS are considered a major rewrite. Revised formatting (margins, font, tabs, etc.) of content is used to create a continuous electronic file, revised numbering of pages and sections and the deletion of unused placeholders, where appropriate, is used to condense and reduce the number of pages in the TS without affecting the technical content. Since the changes to the TS are considered a major rewrite, revision bars are not used. The TS Table of Contents is revised to reflect the remaining applicable sections and new page numbering. These changes are considered administrative and are shown in the marked-up pages (Attachment 2). Note that TS 3/4.14 and 3/4.16 were previously deleted by License Amendment Number 166 (TS 3.16 is marked as not used) and are not discussed further in this proposed change.

License Finding 1.B.	
Current License Finding 1.B. Construction of the Oyster Creek Nuclear Generation Station (Oyster Creek or the facility) has been completed in conformity with Provisional Construction Permit No. CPPR-15, the application, as amended, the provisions of the Act; and the rules and regulations of the Commission.	Proposed License Finding 1.B. ***DELETED***
Basis	
This license finding is proposed for deletion in its entirety. Decommissioning of OCNGS is not dependent on the regulations that govern construction of the facility.	

License Finding 1.C.	
Current License Finding 1.C. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the term of this Renewed Facility Operating License No. DPR-16 on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;	Proposed License Finding 1.C. ***DELETED***

Basis
This license finding is proposed for deletion in its entirety. OCNGS will permanently cease operation and certify that fuel has been permanently removed from the reactor prior to the end of the period of extended operation. Since 10 CFR 50.82(a)(2) prohibits operation of the OCNGS reactor once the certifications described therein are submitted, OCNGS will not operate during the remaining period of extended operation. Decommissioning of OCNGS is not dependent on the requirements of 10 CFR 54 for a renewed license. Therefore, requirements that are unique to a renewed license are not needed.

License Finding 1.D.	
Current License Finding 1.D. The facility will operate in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission (except as exempted from compliance in Section 2.D. below);	Proposed License Finding 1.D. The facility will ***be maintained*** ~~operate~~ in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission ~~(except as exempted from compliance in Section 2.D. below)~~;
Basis	
This license finding is proposed for revision to reflect a more accurate description of the future requirements. Reference to exemptions in Section 2.D is proposed for deletion because Section 2.D is proposed for elimination in its entirety. Since the OCNGS license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR 50.82(a)(2), the removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with the decommissioning plant.	

License Finding 1.E.	
Current License Finding 1.E. There is reasonable assurance (i) that the activities authorized by this operating license 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 rules and regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);	Proposed License Finding 1.E. There is reasonable assurance (i) that the activities authorized by this ~~operating~~ license 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 rules and regulations set forth in 10 CFR Chapter I ~~(except as exempted from compliance in Section 2.D. below)~~;
Basis	
This license finding is proposed for revision to reflect the change from an operating license to a non-operating license. Reference to exemptions in Section 2.D is proposed for deletion because Section 2.D is proposed for elimination in its entirety. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2).	

License Condition 2.B.(1)	
Current License Condition 2.B.(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, to possess, use, and operate Oyster Creek	Proposed License Condition 2.B.(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, to possess~~,~~ ***and*** use~~, and operate~~ Oyster

Nuclear Generation Station at the designated location on the Oyster Creek site in Ocean County, New Jersey, in accordance with the procedures and limitations set forth in this renewed license;	Creek Nuclear Generation Station at the designated location on the Oyster Creek site in Ocean County, New Jersey, in accordance with the procedures and limitations set forth in this renewed license;

Basis
This license condition is proposed for revision to reflect the change from an operating license to being prohibited from operating the reactor pursuant to 10 CFR 50.82(a)(2). As such, the facility would remain authorized to possess the existing spent fuel and use the systems required to support safe fuel storage (e.g., the SFP) during the decommissioning period, in accordance with the specified limitations for storage.

License Condition 2.B.(2)	
Current License Condition 2.B.(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;	Proposed License Condition 2.B.(2) Pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material ***that was used*** as reactor fuel, in accordance with the limitations for storage~~ and amounts required for reactor operation~~, as described in the Updated Final Safety Analysis Report, as supplemented and amended;
Basis	
The proposed change to this license condition removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel. It eliminates the reference to use of the SNM for reactor operations and limits the possession of SNM to SNM "that was used" as reactor fuel at OCNGS. Pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for OCNGS no longer authorizes operation of the reactor. As such, OCNGS has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as OCNGS currently possesses the reactor fuel that was used for the past operations of the reactor.	

License Condition 2.B.(3)	
Current License Condition 2.B.(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;	Proposed License Condition 2.B.(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 ***that were used*** for reactor startup, sealed sources ***that were used*** for ***calibration of*** reactor instrumentation and ***are used in*** radiation monitoring equipment~~ calibration,~~ and as fission detectors in amounts as required;
Basis	
The proposed change to this license condition removes the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup; but retains authorization to possess such sources previously used for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that OCNGS will no longer be authorized to operate and the continued authorization to possess neutron sources that were used for reactor startup is consistent with	

the safe storage of byproduct, source, and SNM. The use of sources for radiation monitoring will continue to be required. Since the OCNGS license will no longer authorize use of the facility for power operation or emplacement or retention of fuel into the reactor vessel, this revision is consistent with the restriction of 10 CFR 50.82(a)(2).

License Condition 2.B.(5)	
Current License Condition 2.B.(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.	Proposed License Condition 2.B.(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~~ ***that were*** produced by the operation of the facility.
Basis	
This license condition is proposed for revision to allow possession of byproduct and SNM that were produced during operation of the reactor, but not allow the separation of material that was produced by operations of the reactor. Since the Part 50 license will no longer authorize operation of the facility pursuant to 10 CFR 50.82(a)(2), this license condition is consistent with the requirements associated with the decommissioning plant.	

License Condition 2.C.(1)	
Current License Condition 2.C.(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.	Proposed License Condition 2.C.(1) ***DELETED***
Basis	
This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2).	

License Condition 2.C.(2)	
Current License Condition 2.C.(2) Technical Specifications The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 291, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.	Proposed License Condition 2.C.(2) Technical Specifications The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ***[XXX]***, are hereby ~~incorporated in the license~~***replaced with the Permanently Defueled Technical Specifications (PDTS)***. Exelon Generation Company shall ~~operate~~ ***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specifications.

Basis
This license condition is proposed for revision to account for the permanently defueled condition of the facility and to incorporate the Permanently Defueled Technical Specifications (PDTS). The paragraph is revised to reflect the nomenclature change to more accurately describe the document. Also changed is the designation from operating to maintaining the facility, which describes the defueled condition in which the OCNGS license no longer allows the use of the facility for power operation as provided in 10 CFR 50.82(a)(2).

License Condition 2.C.(3)	
Current License Condition 2.C.(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.	Proposed License Condition 2.C.(3) ***DELETED***
Basis	
This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, the fire protection program will be revised to take into account the facility conditions and activities during decommissioning. OCNGS will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment. This license condition**,** which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48 with the ability to achieve and maintain safe shut down of the reactor in the event of a fire, will no longer be applicable at OCNGS. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during facility decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. The regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled facility is not needed.	

License Condition 2.C.(4)	
Current License Condition 2.C.(4) Exelon Generation Company shall fully <…> Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP),	Proposed License Condition 2.C.(4) Exelon Generation Company shall fully <…> Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP),

including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by Renewed License Amendment No. 280 and modified by License Amendment No. 288.	including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by Renewed License Amendment No. 280 and modified by License Amendment No***s***. 288 ***and ###***.
Basis	
There are no changes proposed for this License Condition. A change to License Condition 2.C.(4) is under NRC review. A license amendment request (LAR) was submitted on April 10, 2017 (Reference 5), to propose an extension to the Implementation Milestone Regulatory Commitment for the Exelon Cyber Security Plan (CSP) Milestone 8 implementation date as set forth in the Exelon CSP Implementation Schedule.	

License Condition 2.C.(5)	
<u>Current License Condition 2.C.(5)</u> Inspections of core spray spargers, piping and associated components will be performed in accordance with BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," as approved by NRC staff's Final Safety Evaluation Report dated December 2, 1999.[*]	<u>Proposed License Condition 2.C.(5)</u> ***DELETED***
Basis	
This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, inspections of core spray spargers, piping and associated components will no longer be required. However, continued inspections of the core spray spargers, piping and associated components are no longer required for OCNGS because: (a) Exelon has decided to cease power operations of OCNGS, and (b) once fuel has been permanently removed from the OCNGS RPV, there will no longer be refueling outages nor pressurization of the RPV. The core spray system will no longer be required to provide cooling to the reactor in the event of the design basis loss of coolant accident (LOCA). *The NRC is currently reviewing a proposed LAR (Reference 19) for this license condition. The proposed amendment would adopt BWRVIP-18, Revision 2-A, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," which provides relaxed inspection requirements for the Core Spray piping and components currently required to be inspected during the OCNGS OC1R27 refueling outage in Fall 2018. The final wording of this license condition would revise the date of December 2, 1999 to February 22, 2016, and add "and subsequently approved by NRC Letter dated December 21, 2016."	

License Condition 2.C.(6)	
<u>Current License Condition 2.C.(6)</u> Long Range Planning Program – Deleted	<u>Proposed License Condition 2.C.(6)</u> ~~Long Range Planning Program –~~ ***DELETED***
Basis	
This license condition was previously deleted in License Amendment No. 244 (Dated 07/13/2004 ADAMS No. ML041560041). The title is proposed to be removed as part of the RFOL reformatting. This change is considered editorial in nature.	

License Condition 2.C.(7)	
<u>Current License Condition 2.C.(7)</u> Reactor Vessel Integrated Surveillance Program Exelon Generation Company is authorized to revise the Updated Final Safety Analysis Report (UFSAR) to allow implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel Integrated Surveillance Program as the basis for demonstrating compliance with the requirements of Appendix H to Title 10 of the Code of Federal Regulations Part 50, "Reactor Vessel Material Surveillance Program Requirements," as set forth in the licensee's application dated December 20, 2002, and as supplemented on May 30, September 10, and November 3, 2003. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program appropriate for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.	<u>Proposed License Condition 2.C.(7)</u> ***DELETED***

Basis
This license condition was issued concurrent with the Renewed Facility Operating License on April 8, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,"* issued April 2007 (Reference 7). 10 CFR 50 Appendix H requires that the design of the reactor vessel surveillance capsule program and withdrawal schedules meet the requirements in the version of American Society for Testing and Materials (ASTM) Standard Practice E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the RPV was purchased. The rule also requires the licensee to perform capsule testing and to report the test results in accordance with the requirements of ASTM E 185-82 to the extent practicable for the configuration of the test specimen in the RPV surveillance capsules. The requirements in 10 CFR Part 50, Appendix H are only relevant to nuclear plants that are authorized to operate in the reactor-critical operating mode because: (a) this is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process, and (b) the requirements are set in place to provide assurance that the RPV will maintain adequate levels of fracture toughness throughout the operating life of the reactor. This license condition was imposed with the assumption that OCNGS would be operating for an additional 20 years (i.e., to and inclusive of April 9, 2029) and would not be proposing to end power operations of the facility prior to that date. This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Continued implementation of the applicable surveillance capsule testing and reporting

requirements are no longer necessary for OCNGS because: (a) Exelon has decided to cease power operations of OCNGS, and (b) from a fracture toughness perspective, the OCNGS RPV will cease to be exposed to further irradiation by high energy neutrons or subjected to any high thermal stress environments, as induced by operating the RCS at an elevated temperature. Further, 10 CFR 50.60(a) stipulates that reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, no longer need to meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H.

The physical and radiological control of the remaining surveillance capsules that are located in the RPV will be managed in accordance with the applicable radiological control requirements of 10 CFR Part 20 and with any applicable security or physical protection requirements for components in either 10 CFR Part 37 or 10 CFR Part 73. Therefore, the removal, testing, reporting, and storage requirements for reactor vessel surveillance capsules and their test specimens do not need to be implemented further once OCNGS permanently ceases power operations. There will no longer be any need to remove the remaining surveillance capsules from the RPV or perform material testing of the test specimens in those capsules. As such, deletion of this license condition is appropriate. Any corresponding commitments in the OCNGS UFSAR will also be deleted under the provisions of 10 CFR 50.59 upon NRC approval of this proposed license amendment request.

License Condition 2.C.(10)	
<u>Current License Condition 2.C.(10)</u> Upon implementation of Amendment No. 265 adopting TSTF-448, Revision 3, the assessment of CRE habitability as required by Specification 6.22.c.(ii), and the measurement of CRE pressure as required by Specification 6.22.d, shall be considered met. Following implementation: (a) The first performance of the periodic assessment of CRE habitability, Specification 6.22.c.(ii), shall be within 3 years, plus the 9-month allowance of Specification 1.24. (b) The first performance of the periodic measurement of CRE pressure, Specification 6.22.d, shall be within 24 months, plus the 180 days allowed by Specification 1.24, as measured from the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.	<u>Proposed License Condition 2.C.(10)</u> ***DELETED***
Basis	
This license condition is proposed for elimination in its entirety. The proposed change removes the requirements of TSTF-448 that involve assessing the Control Room Envelope (CRE) Habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0. These assessments were completed in accordance with the schedule specified in the license condition. Exelon analyzed the Fuel Handling Accident (FHA) for dose results for the control room (CR)after permanent shutdown. The calculation accounts for radioactive material inventory in the most recently irradiated elements in the SFP after 60 days of decay. For the analysis, Exelon took no credit for CR isolation or filtered recirculation of the control room air. The results of the calculation showed that the dose consequences to occupants in the control room were below acceptable limits. The dose at the control room would be 2.2349E+00 rem, which is less than the 10 CFR 50.67 dose limit of 5 rem. Based on the fact that the dose at the CR is less than the 10 CFR 50.67 dose limit and that no credit was taken for CR isolation or filtered recirculation, the CRE Habitability Program is not required to provide airborne	

radiological protection for the control room operators. This submittal also proposes to remove TS 3.17 for control room ventilation system and TS 6.22 for the CRE Habitability Program. Since TS 3.17 and TS 6.22 are no longer necessary, this license condition is no longer needed; therefore, the proposed deletion of this license condition is acceptable.

License Condition 2.C.(11)	
<u>Current License Condition 2.C.(11)</u> Inspection of Drywell Sand Bed Region The licensee shall perform full scope inspections (as defined in Appendix A of the license renewal safety evaluation report dated March 20, 2007, and summarized in the Updated Final Safety Analysis Report (UFSAR)) of the drywell sand bed region every other refueling outage beginning in the refueling outage prior to April 9, 2009.	<u>Proposed License Condition 2.C.(11)</u> ***DELETED***
Basis	
This license condition was issued concurrent with the Renewed Facility Operating License on April 8, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,"* issued April 2007 (Reference 7). This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, the inspection of Drywell Sand Bed Region will no longer be required. Refueling outages will no longer occur nor will OCNGS operate during the remaining period of extended operation (ending April 9, 2029) and this activities that is unique to the renewed license is not necessary. Decommissioning of OCNGS is not dependent on the requirements of 10 CFR 54, *"Requirements for Renewal of Operating Licenses for Nuclear Power Plants,"* for a renewed license.	

License Condition 2.C.(12)	
<u>Current License Condition 2.C.(12)</u> Drywell Trenches The licensee shall monitor the drywell trenches (as defined in Appendix A of the license renewal safety evaluation report dated March 20, 2007, and summarized in the UFSAR) every refueling outage to identify and eliminate the sources of water and shall receive NRC approval prior to restoring the trenches to their original design configuration.	<u>Proposed License Condition 2.C.(12)</u> ***DELETED***
Basis	
This license condition was issued concurrent with the Renewed Facility Operating License on April 8, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,"* issued April 2007 (Reference 7). This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, monitoring the two trenches for the presence of water during refueling outages	

will no longer be necessary. There will no longer be refueling outages nor need for the drywell shell (primary containment). Decommissioning of OCNGS is not dependent on the requirements of 10 CFR 54, *"Requirements for Renewal of Operating Licenses for Nuclear Power Plants,"* for a renewed license.

License Condition 2.C.(13)	
Current License Condition 2.C.(13) Engineering Study of Refueling Cavity Liner The licensee shall perform an engineering study prior to April 9, 2009 to identify options to eliminate or reduce the leakage in the facility cavity liner.	Proposed License Condition 2.C.(13) ***DELETED***
Basis	
This license condition was issued concurrent with the Renewed Facility Operating License on April 8, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,"* issued April 2007 (Reference 7). This license condition is a one-time requirement that has been completed and is proposed for deletion in its entirety. On March 27, 2009, the NRC completed a License Renewal Follow-up Inspection at OCNGS (Reference 20). The NRC did not identify any significant problems or concerns. Based on the conclusion of NRC's review, this license condition has been completed in its entirety and may be eliminated. Decommissioning of OCNGS is not dependent on the requirements of 10 CFR 54, *"Requirements for Renewal of Operating Licenses for Nuclear Power Plants,"* for a renewed license.	

License Condition 2.C.(14)	
Current License Condition 2.C.(14) Three-Dimensional Finite-Element Analysis of Drywell Shell The licensee shall perform a three-dimensional finite-element analysis of the drywell shell and shall provide to the NRC staff a report of the results prior to April 9, 2009.	Proposed License Condition 2.C.(14) ***DELETED***
Basis	
This license condition was issued concurrent with the Renewed Facility Operating License on April 8, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,"* issued April 2007 (Reference 7). This license condition is a one-time aging management program activity that was added to the Renewed License. The NRC reviewed Exelon's Three-Dimensional analysis summary report and concluded that the Exelon fulfilled its commitment. Based on the NRC's review (Reference 21), this license condition has been completed in its entirety and may be eliminated. Decommissioning of OCNGS is not dependent on the requirements of 10 CFR 54, *"Requirements for Renewal of Operating Licenses for Nuclear Power Plants,"* for a renewed license.	

License Condition 2.C.(15)	
<u>Current License Condition 2.C.(15)</u> UFSAR Supplement Changes The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4), as modified by an exemption granted by letter dated July 7, 2004 (ADAMS Accession No. ML041340673), following the issuance of this renewed operating license. Until that update is complete, Exelon Generation Company may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Exelon Generation Company evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.	<u>Proposed License Condition 2.C.(15)</u> ***DELETED***

Basis
This license condition was issued concurrent with the Renewed Facility Operating License on April 8, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1875, *"Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,"* issued April 2007 (Reference 7). This license condition is a one-time requirement to update the UFSAR to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update as required by 10 CFR 50.71(e) and allows changes to be made to that supplement under the provisions of 50.59 until the UFSAR update is completed. OCNGS UFSAR, Revision 16, which included the supplement (Appendix A) for the License Renewal Application (LRA) (ECR 09-00141), has been submitted to the NRC (Reference 22). This action satisfies the requirements of OCNGS Renewed Facility Operating License Condition 2.C.(15). Since the UFSAR supplement was submitted to the NRC and the license renewal commitments have been incorporated in the UFSAR, these actions satisfy the requirements of OCNGS Renewed Facility Operating License Condition 2.C.(15). This license condition is proposed for deletion in its entirety. Decommissioning of OCNGS is not dependent on the requirements of 10 CFR 54, *"Requirements for Renewal of Operating Licenses for Nuclear Power Plants,"* for a renewed license.

License Condition 2.D.	
<u>Current License Condition 2.D.</u> The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and	<u>Proposed License Condition 2.D.</u> ***DELETED***

sent to the licensee in letters dated March 24, 1986 and June 25,1990. The facility has also been granted certain exemptions from the requirements of Section III.J of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to emergency lighting that shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto. This exemption was granted and sent to the licensee in a letter dated February 12, 1990. In addition, the facility has been granted certain exemptions from Section 55.45(b)(2)(iii) and (iv) of 10 CFR Part 55, "Operators' Licenses." These sections contain requirements related to site-specific simulator certification and require that operating tests will not be administered on other than a certified or an approved simulation facility after May 26, 1991. These exemptions were granted and sent to the licensee in a letter dated March 25, 1991. These exemptions granted pursuant to 10 CFR 50.12 are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.	

Basis

This license condition is proposed for deletion in its entirety. This license condition documents specific exemption from 10 CFR Part 50 and 10 CFR Part 55 as approved by the NRC. Specifically, exemption from the requirements of 10 CFR Part 50, Appendix R, Section III.G and III.J; and exemptions from Section 55.45(b)(2)(iii) and (iv) of 10 CFR Part 55, "*Operators' Licenses*."

The requirements of Appendix R required to mitigate the consequences of design basis accidents under post-fire conditions, limit fire damage to systems required to achieve, and maintain safe shutdown conditions. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, the requirements of Appendix R will no longer apply. Therefore, OCNGS will no longer need these exemptions to Appendix R. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. The regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled facility is not needed. The fire protection program will be revised to take into account the facility conditions and activities during decommissioning. OCNGS will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment.

The sections exempted in 10 CFR Part 55 are no longer contained in the requirements and are proposed to be administratively removed. This action is editorial in nature. Further, once OCNGS has certified to the NRC the permanent cessation of operation and removal of fuel from the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the OCNGS license will no longer authorize operation of the reactor or emplacement

or retention of fuel into the reactor vessel, and OCNGS will no longer be required to have operators licensed pursuant to 10 CFR Part 55.

License Condition 2.E, 3.A - K	
2.E. Deleted 3.A through 3.C Deleted. 3.D through 3.K Deleted	2.E. ***DELETED*** ~~Deleted~~ 3.A through 3.C ***DELETED*** ~~Deleted.~~ 3.D through 3.K ***DELETED*** ~~Deleted~~
Basis	
Proposed format change for the term "deleted." This change is editorial in nature.	

License Condition 3. Sales and License Transfer Conditions	
M. At the time of the closing of the transfer of Oyster Creek, and the respective license from AmerGen Energy Company, LLC (AmerGen) to Exelon Generation Company, AmerGen shall transfer to Exelon Generation Company ownership and control of AmerGen Oyster Creek NQF, LLC, and AmerGen Consolidation, LLC shall be merged into Exelon Generation Consolidation, LLC. Also at the time of the closing, decommissioning funding assurance provided by Exelon Generation Company, using an additional method allowed under 10 CFR 50.75 if necessary, must be equal to or greater than the minimum amount calculated on that date pursuant to, and required by 10 CFR 50.75 for Oyster Creek. Furthermore, funds dedicated for Oyster Creek prior to closing shall remain dedicated to Oyster Creek following the closing. The name of AmerGen Oyster Creek NQF, LLC shall be changed to Exelon Generation Oyster Creek NQF, LLC at the time of the closing.	M. ***DELETED***
Basis	
This license condition is proposed for deletion in its entirety. This license condition eliminates references to AmerGen Energy Company, LLC, and replaces them with references to Exelon Generation Company, LLC, to reflect the results of the license transfer. AmerGen transferred to Exelon Generation Company ownership and control of AmerGen Oyster Creek NQF, LLC, and AmerGen Consolidation, LLC merged into Exelon Generation Consolidation, LLC. On December 23, 2008, the NRC approved the transfer of license and ownership of OCNGS to Exelon (Reference 22). The name of AmerGen Oyster Creek NQF, LLC was changed to Exelon Generation Oyster Creek NQF, LLC at the time of the closing. In a letter dated March 31, 2009, Exelon reported to the NRC that the decommissioning trust agreements for OCNGS had been modified to reflect the change in license from AmerGen Energy Co. LLC to Exelon (Reference 15). The requirements of this license condition have been completed; therefore, this license condition may be eliminated.	

License Condition 4.	
<u>Current License Condition 4.</u> This license is effective as of the date of issuance and shall expire at midnight on April 9, 2029.	<u>Proposed License Condition 4.</u> This license is effective as of the date of issuance and ~~shall expire at midnight on April 9, 2029~~***is effective until the Commission notifies the licensee in writing that the license is terminated.***
Basis	
The proposed change modifies this license condition to reflect the permanently defueled condition of the facility. Once OCNGS has permanently ceased operation and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2). This license condition is being revised to conform with 10 CFR 50.51, *"Continuation of license,"* in that the license authorizes ownership and possession by Exelon until the Commission notifies the licensee in writing that the license is terminated.	

TS Section 1 – Definitions	
TS Section 1 "Definitions," contains defined terms that are applicable to an operating plant throughout the TS and TS Bases. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The revision to the definitions identified below align with the permanently shutdown and defueled reactor conditions. Many of the definitions have been proposed for deletion since they are relevant to an operating reactor and are no longer used in the TS. The standard convention of indicating the defined term in ALL CAPITAL LETTERS throughout the TS has been adopted in the PDTS.	
Definitions Maintained	**Basis**
1.52 <u>CERTIFIED FUEL HANDLER</u> A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.	The definition for Certified Fuel Handler was approved by the NRC in TS Amendment No. 290 (Reference 3). The definition is proposed to be renumbered from 1.52 to 1.2 to place it in alphabetic order with the remaining TS definitions. This action is editorial in nature.
1.53 <u>NON-CERTIFIED OPERATOR</u> A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.	The definition for Non-Certified Operator was approved by the NRC in TS Amendment No. 290 (Reference 3). The definition is proposed to be renumbered from 1.53 to 1.3 to place it in alphabetic order with the remaining TS definitions. This action is editorial in nature.
Proposed Definitions - Added	**Basis for Addition**
<u>ACTIONS</u> ***ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.***	The definition for "Actions" is being added in order to clarify a term used in remaining TS sections. The definition is based on the definition in NUREG-1433, *"Standard Technical Specifications General Electric BWR/4 Plants"* (Reference 7). The definition is proposed to be numbered 1.1 to place it in alphabetic order with the remaining TS definitions. This action is editorial in nature.

Proposed Definitions - Deleted	Basis for Deletion
1.1 OPERABLE-OPERABILITY A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling of seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). A verification of operability is an administrative check, by examination of appropriate plant records (logs, surveillance test records) to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine operability.	This definition is proposed for deletion since the term is not used in any PDTS specification. There are no systems or components required to be operable in the PDTS.
1.2 OPERATING Operating means that a system or component is performing its required function.	This definition is proposed for deletion since the term is not used in any PDTS specification. There are no systems or components required to be operable in the PDTS.
1.3 POWER OPERATION Power operation is any operation when the reactor is in the startup mode or run mode except when primary containment integrity is not required.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.4 STARTUP MODE The reactor is in the startup mode when the reactor mode switch is in the startup mode position. In this mode, the reactor protection system scram trips initiated by condenser low vacuum and main steam line isolation valve closure are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure is bypassed; the IRM trips for rod block and scram are operable; and the SRM trips for rod block are operable.	This definition is proposed for deletion since the term is not used in any PDTS specification. Operating Modes are defined for operating or refueling conditions and do not apply to a facility in the permanently defueled condition.
1.5 RUN MODE The reactor is in the run mode when the reactor mode switch is in the run mode position. In this mode, the reactor protection system is energized with APRM protection and the control rod withdrawal interlocks are in service.	This definition is proposed for deletion since the term is not used in any PDTS specification. Operating Modes are defined for operating or refueling conditions and do not apply to a facility in the permanently defueled condition.

1.6 SHUTDOWN CONDITION The reactor is in the SHUTDOWN CONDITION when there is fuel in the reactor vessel, the reactor is subcritical, all operable control rods are fully inserted, and the mode switch is in the shutdown mode position. In this position, a control rod block is initiated.	This definition is proposed for deletion since the term is not used in any PDTS specification. Operating Modes are defined for operating or refueling conditions and do not apply to a facility in the permanently defueled condition.
1.7 COLD SHUTDOWN CONDITION The reactor is in the COLD SHUTDOWN CONDITION when the reactor is in the SHUTDOWN CONDITION, and (except during REACTOR VESSEL PRESSURE TESTING), the reactor coolant system is maintained at less than 212°F and vented.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. Reactor Vessel Pressure Testing will no longer be required as discussed below in the Basis for definition 1.39. It does not apply to a facility in the permanently defueled condition.
1.8 PLACE IN SHUTDOWN CONDITION Proceed with and maintain an uninterrupted normal plant shutdown operation until the SHUTDOWN CONDITION is met.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.9 PLACE IN COLD SHUTDOWN CONDITION Proceed with and maintain an uninterrupted normal plant shutdown operation until the COLD SHUTDOWN CONDITION is met.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.10 PLACE IN ISOLATED CONDITION Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.11 REFUEL MODE The reactor is in the REFUEL MODE when the reactor mode switch is in the REFUEL MODE position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. Operating Modes are defined for operating or refueling conditions and do not apply to a facility in the permanently defueled condition.
1.12 REFUELING OUTAGE For the purpose of designating frequency of testing and surveillance, a REFUELING OUTAGE shall mean a regularly scheduled REFUELING OUTAGE. Following the first REFUELING OUTAGE, successive tests or surveillances shall be performed at least once per 24 months.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. There will be no more refueling outages once OCNGS has certified that fuel has been permanently removed from the reactor.
1.13 PRIMARY CONTAINMENT INTEGRITY PRIMARY CONTAINMENT INTEGRITY means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied:	This definition is proposed for deletion since the term is not used in any PDTS specification. In the permanently defueled condition there will be no DBAs

A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed. B. At least one door in the airlock is closed and sealed. C. All automatic primary containment isolation valves are OPERABLE or the affected penetration is isolated. D. All blind flanges and manways are closed.	for which primary containment integrity will be required to mitigate the consequences.
1.14 <u>SECONDARY CONTAINMENT INTEGRITY</u> Secondary containment integrity means that the reactor building is closed and the following conditions are met: A. At least one door at each access opening is closed. (Note: Momentary opening and closing of the trunnion room door does not constitute a loss of secondary containment integrity. In COLD SHUTDOWN CONDITION or REFUEL MODE, the trunnion room door may remain open provided the trunnion room is isolated from the secondary containment through the reactor building walls, penetrations and either the inboard or outboard valves to the main steam and feedwater piping being secured in the closed position.) B. The standby gas treatment system is operable. C. All automatic secondary containment isolation valves are operable or are secured in the closed position.	This definition is proposed for deletion since the term is not used in any PDTS specification. Exelon has completed an analysis of a FHA in the SFP using the guidelines detailed in Regulatory Guide 1.183. No other credible accidents rely on secondary containment. The analysis demonstrates that radiological doses at the exclusion area boundary, low population zone and in the control room from a FHA after 60 days following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system.
1.15 Deleted	This placeholder is proposed to be removed due to the elimination of other definitions. This action is editorial in nature.
1.16 <u>RATED FLUX</u> Rated flux is the neutron flux that corresponds to a steady state power level of 1930 MW(t). Use of the term 100 percent also refers to the 1930 thermal megawatt power level.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.17 <u>REACTOR THERMAL POWER-TO-WATER</u> Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.

1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS A. INSTRUMENT CHANNEL An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel. B. TRIP SYSTEM A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).	These definitions are proposed for deletion since the terms are not used in any PDTS specification. There are no instruments or logic systems credited in the analysis of the accidents that remain credible (the FHA in the SFP).
1.19 INSTRUMENT 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	These definitions are proposed for deletion since the terms are not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible (the FHA in the SFP).

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.	This definition is proposed for deletion since the term is not used in any PDTS specification.
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.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is no longer applicable since fuel will be permanently removed from the reactor core.
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.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.23 Deleted	This placeholder is proposed to be removed due to the elimination of other definitions. This action is editorial in nature.
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. 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	This definition is being reformatted, revised, and relocated to Section 3/4.0 "Limiting Conditions for Operation and Surveillance Requirement Applicability," as SR 4.0.4. The proposed change will ensure the appropriate requirements for the 25% grace period are maintained (see discussion of SR 4.0.4). The portion of the definition with respect to modes is being deleted due to the eliminated of reactor modes. The remainder of this definition is being relocated and revised to the TS Basis for SR 4.0.4, as shown in Attachment 2. The wording is consistent with NUREG-1433 which has been modified to reflect a permanently shutdown condition.

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, Pa = 35 psig.	This definition is proposed for deletion since the term is not used in any PDTS specification. The requirements of 10 CFR 50 Appendix J are no longer applicable to a facility that has submitted the certificates required under 10 CFR 50.82(a)(1).
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.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
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).	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.28 FRACTION OF RATED POWER (FRP) The FRACTION OF RATED POWER is the ratio of core THERMAL POWER to RATED THERMAL POWER.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.29 TOP OF ACTIVE FUEL (TAF) 353.3 inches above vessel zero.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.30 REPORTABLE EVENT	This definition is proposed for deletion since the term is not used in any PDTS specification. The term is

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.	defined and codified in the applicable regulations (e.g. 10 CFR 50.72 and 10 CFR 50.73); therefore, the definition need not be repeated in the PDTS. Administrative Controls TS 6.6, "Reportable Events Action," was deleted in TS Amendment No. 290 (Reference 3).
1.31 <u>IDENTIFIED LEAKAGE</u> IDENTIFIED LEAKAGE is that leakage which is collected in the primary containment equipment drain tank and eventually transferred to radwaste for processing.	This definition is proposed for deletion since the term is not used in any PDTS specification. Refer to discussion for elimination of TS 3.3.D below.
1.32 <u>UNIDENTIFIED LEAKAGE</u> UNIDENTIFIED LEAKAGE is all measured leakage that is other than identified leakage.	This definition is proposed for deletion since the term is not used in any PDTS specification. Refer to discussion for elimination of TS 3.3.D below.
1.33 <u>PROCESS CONTROL PLAN</u> The PROCESS CONTROL PLAN shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.	This definition is proposed for deletion since the term is not used in any PDTS specification.
1.34 <u>AUGMENTED OFFGAS SYSTEM (AOG)</u> The AUGMENTED OFFGAS SYSTEM is a system designed and installed to holdup and/or process radioactive gases from the main condenser offgas system for the purpose of reducing the radioactive material content of the gases before release to the environs.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.35 <u>MEMBER OF THE PUBLIC</u> A MEMBER OF THE PUBLIC is a person who is not occupationally associated with Exelon Generation Company, LLC and who does not normally frequent the Oyster Creek Nuclear Generating Station site. The category does not include contractors, contractor employees, vendors, or persons who enter the site to make deliveries, to service equipment, work on the site, or for other purposes associated with plant functions.	This definition is proposed for deletion since the term is defined in 10 CFR 20.1003; therefore, the definition need not be repeated in the PDTS.
1.36 <u>OFFSITE DOSE CALCULATION MANUAL (ODCM)</u> The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/trip	This definition has been located to the ODCM.

Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4; and (2) descriptions of the information that should be included in the Annual Radioactive Effluent Release Report AND Annual Radiological Environmental Operating Report required by Specifications 6.9.1.d and 6.9.1.e, respectively.	
1.37 PURGE PURGE OR PURGING is the controlled process of discharging air or gas from a confinement and replacing it with air or gas.	This definition is proposed for deletion since the term is not used in any PDTS specification. There are no purge activities credited in the analyses of the accidents that remain credible.
1.38 SITE BOUNDARY The SITE BOUNDARY is the perimeter line around the OCNGS beyond which the land is neither owned, leased nor otherwise subject to control by Exelon Generation Company, LLC (ref. ODCM). The area outside the SITE BOUNDARY is termed OFFSITE or UNRESTRICTED AREA.	This definition is proposed for deletion since the term is defined in 10 CFR 20.1003; therefore, the definition need not be repeated in the PDTS.
1.39 REACTOR VESSEL PRESSURE TESTING System pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely water solid, core not critical and section 3.2.A satisfied.	This definition is proposed for deletion since the term is not used in any PDTS specification. Once the reactor vessel is permanently defueled, the applicable requirements for RPV testing will no longer apply because the reactor coolant pressure boundary will no longer be used as a fission product barrier.
1.40 SUBSTANTIVE CHANGES SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Example of non-substantive changes are: (1) correcting spelling, (2) adding (but not deleting) sign-off spaces, (3) blocking in notes, cautions, etc, (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications, and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.	This definition is proposed for deletion since the term is not used in any PDTS specification.
1.41 DOSE EQUIVALENT I-131 DOSE EQUIVALENT I-131 shall be that concentration of I-131 microcuries per gram which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 or Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluences for the Purpose of	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is used in current TS 3.6.A to express the specific activity limit from a mixture of iodine isotopes and contained in primary coolant. TS 3.6.A is proposed for deletion in the PDTS. The specific activity limit is used as the basis in accident analysis involving primary coolant releases. Since accident conditions associated with the reactor coolant system will no longer apply to the permanently shut down and

Evaluating Compliance with 10 CFR Par 40 Appendix I."	defueled facility, the definition is no longer meaningful.
1.42 <u>AVERAGE PLANAR LINEAR HEAT GENERATION RATE</u> The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.43 <u>CORE OPERATING LIMITS REPORT</u> The Oyster Creek CORE OPERATING LIMITS REPORT (COLR) is the document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.f. Plant operation within these operating limits is addressed in individual specifications.	This definition is proposed for deletion since the term is not used in any PDTS specification. Specification 6.9.1.f that requires the COLR has been deleted from PDTS in TS Amendment No. 290 (Reference 3).
1.44 <u>LOCAL LINEAR HEAT GENERATION RATE</u> The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) shall be applicable to a specific planar height and is equal to the AVERAGE PLANAR LINEAR GENERATION RATE (APLHGR) at the specified height multiplied by the local peaking factor at that height.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.45 <u>SHUTDOWN MARGIN (SDM)</u> SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.46 <u>IDLE RECIRCULATION LOOP</u> A recirculation loop is idle when its discharge valve is in the closed position and its discharge bypass valve and suction valve are in the open position.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.47 <u>ISOLATED RECIRCULATION LOOP</u>	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain

A recirculation loop is fully isolated when the suction valve, discharge valve and discharge bypass valve are in the closed position.	fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.48 <u>OPERATIONAL CONDITION</u> The reactor plant operational status as to criticality, reactor mode switch position, reactor coolant temperature, and/or specific system status. These conditions consist of POWER OPERATION, STARTUP MODE, SHUTDOWN CONDITION, COLD SHUTDOWN CONDITION, and REFUEL MODE. A change or entry into an operating condition is Signified by movement of the reactor mode switch or a change in reactor coolant Temperature from <212°F to ≥212ºF.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.49 <u>RATED THERMAL POWER (RTP)</u> RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.50 <u>THERMAL POWER</u> THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	This definition is proposed for deletion since the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
1.51 <u>PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)</u> The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.23.	This definition is proposed for deletion since the term is not used in any PDTS specification. This report does not apply to a facility in the permanently defueled condition. TS 6.23 is being proposed for deletion from the PDTS.

TS SECTION 2 – SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

TS Section 2 "Safety Limits and Limiting Safety System Settings," contains "safety limits" and "limiting safety system settings" to establish limits on important process variables to assure the integrity of the fuel cladding and the Reactor Coolant System (RCS) in all Modes of operation.

Pursuant to 10 CFR 50.36(c)(1), safety limits are limiting parameters necessary to protect the physical barriers that guard against uncontrolled release of radioactivity from a nuclear reactor. The Safety Limits established in TS 2.1 and 2.2 protect the integrity of the fuel cladding and reactor coolant system barriers, respectively. Limiting safety system settings in TS 2.3 are values of various parameters associated with the Nuclear Steam Supply System at which automatic protective action is needed during normal operations or anticipated transients to prevent exceeding a safety limit.

This section is being proposed for deletion in its entirety, since the safety limits do not apply to a reactor that is in a permanently defueled condition. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). These specifications do not apply to the safe storage and handling of spent fuel in the spent fuel pool (SFP).

Current OCNGS TS	Basis for Deletion
TS 2.1, Safety Limits – Fuel Cladding Integrity	TS 2.1 provides safety limits that prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. This specification is applicable during all Modes of reactor operation with fuel in the reactor vessel. Pursuant to 10 CFR 50.82(a)(2), the facility license for OCNGS will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Since the safety limits and limiting safety system settings apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limits and limiting safety system settings are proposed for deletion.
TS 2.2 Safety Limits – Reactor Coolant System Pressure	TS 2.2 provide safety limits that prevent potential damage to the reactor coolant pressure boundary (RCPB) from overpressure that could result in the uncontrolled release of fission products to the containment atmosphere. This specification is applicable during Power Operation and Operational Conditions with fuel in the reactor vessel. Pursuant to 10 CFR 50.82(a)(2), the facility license for OCNGS will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Since the safety limits and limiting safety system settings apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limits and limiting safety system settings are proposed for deletion.
TS 2.3, Limiting Safety System Settings	TS 2.3 provide limiting safety system settings to ensure the safety limits in TS 2.1 and 2.2 are not exceeded. The TS establishes the trip settings for automatic protective devices that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the facility during Normal Power Operation or Operational Transients conditions. Pursuant to 10 CFR 50.82(a)(2), the facility license for OCNGS will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Since the safety limits and limiting safety system settings apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limits and limiting safety system settings are proposed for deletion.

TS SECTION 3 – LIMITING CONDITION FOR OPERATION

TS Section 3 of the current OCNGS TS contains the Limiting Conditions for Operation (LCO). In accordance with 10 CFR 50.36(c)(2), LCOs specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The LCOs typically place restrictions on availability of safety equipment needed to prevent or mitigate a postulated Design Basis Accident (DBA), or on process variables necessary to preserve the initial conditions assumed in the safety analyses of postulated DBAs. 10 CFR 50.36(c)(2)(ii) defines four criteria for establishing LCOs (see Section 3.1). Associated surveillance requirements help to ensure that specified equipment and parameters are maintained within the limits specified in the LCOs.

As discussed previously, only one postulated DBA (e.g., the FHA in the SFP) remains applicable to OCNGS with the reactor in the permanently defueled state. As a result, most of the LCOs and accompanying surveillance requirements contained in the OCNGS TS are determined to be inappropriate for retention in the proposed PDTS.

Due to the reduced number of LCOs and Surveillance Requirements, OCNGS proposes to combine the LCOs (TS Section 3) with the corresponding Surveillance Requirements (TS Section 4) This format will allow the Surveillance Requirements to be more readily associated with the corresponding LCO. The LCOs and combined Surveillance Requirements (SR) sections will be designated with notation 3/4.#. The proposed format to the LCOs is shown in Attachment 2.

The list below contains a comparison between the provisions of the current OCNGS TS and the proposed PDTS. Each subsection of OCNGS TS Section 3 is discussed in more detail in the tables below.

Current OCNGS TS		Proposed PDTS
3.0	Limiting Conditions for Operation (General)	3/***4***.0 Limiting Conditions for Operation ~~(General)~~***and Surveillance Requirement Applicability***
3.1	Protective Instrumentation	Not Applicable (Proposed New ***3/4.1 Spent Fuel Storage***)
3.2	Reactivity Control	Not Applicable (Proposed ***3/4.2 Radioactive Liquid Storage***)
3.3	Reactor Coolant	Not Applicable
3.4	Emergency Cooling	Not Applicable
3.5	Containment	Not Applicable
3.6	Radioactive Effluents	Remaining LCO 3.6.C Renumbered as 3/4.2
3.7	Auxiliary Electrical Power	Not Applicable
3.8	Isolation Condenser	Not Applicable
3.9	Refueling	Not Applicable
3.10	Core Limits	Not Applicable
3.11	Not Used	
3.12	Alternate Shutdown Monitoring Instrumentation	Not Applicable
3.13	Accident Monitoring Instrumentation	Not Applicable
3.14	Deleted	
3.15	Explosive Gas Monitoring Instrumentation	Not Applicable
3.16	Not Used	
3.17	Control Room Heating, Ventilation and Air Conditioning System	Not Applicable

TS SECTION 3.0 – LIMITING CONDITION FOR OPERATION (GENERAL)

TS Section 3.0 "Limiting Condition for Operation (GENERAL)," contains the general requirements applicable to all LCOs and applies at all times unless otherwise stated in TSs. Due to the limited number of LCOs in the proposed PDTS, a number of the OCNGS TS provisions in this section are no longer necessary or applicable to the OCNGS facility as indicated in the following table. LCOs 3.0.1 and 3.0.2 are being proposed for addition in the PDTS. These LCOs are based on NUREG-1433, *"Standard Technical Specifications General Electric BWR/4 Plants"* (Reference 8) and Draft NUREG-1625, *"Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants"* (Reference 13), which have been modified to reflect the permanently defueled condition.

It is proposed to combine TS Section 3.0 and Section 4.0 for TS LCOs (GENERAL) and Surveillance Requirement Applicability into a common specification. The TS Section will be retitled TS Section ***3/4.0, LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENT APPLICABILITY***. The proposed format for Section 3/4.0 is shown in Attachment 2.

Current OCNGS TS	Basis for Deletion
LCO 3.0.A In the event Limiting Conditions for Operation (LCOs) and/or associated action	TS 3.0.A provides 30 hours for corrective measures to be completed that allow continued

requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable. Exceptions to the requirements shall be stated in the individual specifications.	operation when the requirements of an LCO are exceeded. The specification is applicable during Operational Conditions and delineates the action to be taken for circumstances not directly provided for in the system LCOs and whose occurrence violates the intent of the specification. This TS LCO is proposed for deletion in PDTS. Since 10 CFR 50.82(a)(2) prohibits power operation of the plant or placing fuel in the reactor vessel, TS 3.0.A will no longer be applicable.
LCO 3.0.B When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of applicable LCOs., provided (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in COLD SHUTDOWN within the following 30 hours or within the time specified in the applicable specification. This specification is not applicable in COLD SHUTDOWN or the REFUEL MODE.	TS 3.0.B delineates what additional conditions must be satisfied to permit operation of systems, subsystems, trains, components, or devices to continue, consistent with the specifications for power sources, when a normal or emergency power source is not operable. This TS LCO is proposed for deletion in PDTS since 10 CFR 50.82(a)(2) prohibits power operation or placing fuel in the reactor vessel. Because there are no systems required for the permanently defueled condition that have emergency power supply requirements, this specification is no longer applicable.
LCO 3.0.C When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made: 1. When the associated LCO requirement permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or 2. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the applicability, and the establishment of risk management actions, if appropriate; exceptions to this specification are stated in the individual Specifications, or 3. When an allowance is stated in the individual value, parameter, or other Specification. This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.	TS 3.0.C establishes limitations on changes in Operational Conditions or other specified conditions in the Applicability when an LCO is not met. It allows placing the plant in an Operational Condition or other specified condition stated in the Applicability when plant conditions are such that the requirements of the LCO would not be met, in accordance with LCOs 3.0.C.1, 3.0.C.2, or 3.0.C.3. This TS LCO is proposed for deletion in PDTS since all actions in the PDTS have a completion time of "Immediately" or "As soon as reasonably achievable." This makes LCO 3.0.C unnecessary. Further, 10 CFR 50.82(a)(2) prohibits power operation or placing fuel in the reactor vessel, the reference to Modes or operational conditions will no longer be relevant.

Proposed TS Section ***3/4.0*** ***<u>LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENT APPLICABILITY</u>***	
Proposed OCNGS TS	**Basis for Addition**
LCO 3.0.1 ***LCOs shall be met during the specified conditions in the TS, except as provided in TS 3.0.2.***	TS 3.0.1 is proposed as an addition to the PDTS. The specification establishes the applicability statement within each individual TS as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the applicability statement of each specification). This statement is based on LCO 3.0.1 in NUREG-1433 (Reference 8), and Draft NUREG-1625 (Reference 13).
LCO 3.0.2 ***Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.*** ***If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.***	TS 3.0.2 is proposed as an addition to the PDTS. The specification establishes that upon discovery of a failure to meet an LCO, the associated action shall be met. The completion time of each required action for an action condition is applicable from the point in time that an action condition is entered. The required actions establish those remedial measures that must be taken within specified completion times when the requirement of an LCO are not met. This statement is based on LCO 3.0.2 in NUREG-1433 (Reference 8), and Draft NUREG-1625 (Reference 13).
SR 4.0.1 ***Surveillance requirements shall be met during the specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in 4.0.4. Surveillances do not have to be performed on variables outside specified limits.***	SR 4.0.1 is relocated from current TS Section 4.0 "Surveillance Requirement Applicability" to immediately follow the LCO statement in proposed PDTS Section 3/4.0. See current TS LCO 4.0.1 for justification for proposed wording.
SR 4.0.2 ***If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery,***	SR 4.0.2 is relocated from current TS Section 4.0 "Surveillance Requirement Applicability" to immediately follow the LCO statement in proposed PDTS Section 3/4.0. See current TS LCO 4.0.2 for justification for proposed wording.

up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed. ***If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.*** ***When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.***	
SR 4.0.3 ***Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillance has been met within its specified frequency, except as provided by 4.0.2.*** ***This provision shall not prevent entry into other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.***	SR 4.0.3 is relocated from TS Section 4.0 "Surveillance Requirement Applicability." See current TS LCO 4.0.3 for justification.
SR 4.0.4 ***The specified frequency for each SR is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.***	SR 4.0.4 is being proposed for addition to the Surveillance Requirement Applicability section. This specification is based upon the OCNGS TS Definition for "Surveillance Requirement," which states, in part, "Each surveillance requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval." The wording of the proposed specification is from NUREG-1433 (Reference 8) and Draft NUREG-1625 (Reference 13), except that it is modified for a facility in permanently defueled condition. There is no change in intent for this statement and the OCNGS TS definition. Both statements provide an allowance for extending the frequency for performance of a SR to 1.25 times the interval specified in the frequency to facilitate scheduling or

	unforeseen problems that may prevent performance during normal intervals.

TS Section ~~3.1 – PROTECTIVE INSTRUMENTATION~~*3/4.1 - SPENT FUEL STORAGE*

TS Section 3.1 "Protective Instrumentation," contains LCOs to assure the operability of protective instrumentation. The LCOs are related to plant instrumentation that performs protective and monitoring functions to ensure safe operation of the reactor and to mitigate the effects of reactor related design basis accidents (DBA). The table below describes the specifications in this section.

The current content of this section is being proposed for deletion in its entirety and will be replaced with a proposed new TS regarding spent fuel storage. Currently, these specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.1 will not be required and these LCOs (and associated surveillances requirements (SR) in Section 4) will not apply in a permanently defueled condition.

The proposed change renumbers and retitles TS Section 3.1 to TS Section 3/4.1 "Spent Fuel Storage" and adds a new specification to address operability requirements for the SFP level. There are currently no TS LCOs related to spent fuel storage other than the design specifications in TS Section 5 Design Features (TS 5.3)(proposed to be renumbered TS 5.2).

Current OCNGS TS	Basis for Deletion
TS 3.1.A - Operating Requirements for Plant Protective Instrumentation given in Table 3.1.1.	This specification provides the operability requirements for Protection Instrumentation specified in Table 3.1.1. Table 3.1.1 defines, for each protective function, the minimum number of operable instrument channels for an operable trip system for the various functions specified; the required mode of operation; and the specified Action Required when the specified LCO cannot be met. Each section of Table 3.1.1 is discussed below. The reactor protection system was included in the OCNGS TS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.1.A, including Table 3.1.1, is proposed for deletion in PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Thus, there will no longer be a need for Protective Instrumentation to protect the reactor core. There are no Modes of Operation once the reactor is defueled.
TS 3.1.B - Average Power Range Monitor Operability	This specification defines the minimum number of Average Power Range Monitor (APRM) channel inputs required to permit accurate average core power monitoring. TS 3.1.B is proposed for deletion in PDTS. Since OCNGS will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted, there will no longer be a need for average core power monitoring.
TS 3.1.C - Local Power Range Monitors (LPRMs) and Traversing In-Core Probes (TIPs)	This specification defines the distribution of the operable chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. This specification provides the requirements for LPRM assemblies and provisions for using a Travelling In-Core Probe

	(TIP) chamber may be used as an APRM input to meet the criteria of 3.1.B and 3.1.C.1.. TS 3.1.C is proposed for deletion in PDTS. Since OCNGS will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted, there will no longer be a need for average or local core power monitoring.
Table 3.1.1, Section A, Scram	The reactor protection system (RPS) was designed to shut down the reactor when one or more parameters exceeded specified limits to prevent or mitigate the consequences of postulated accidents that could result in damage to the reactor fuel cladding or the reactor coolant system pressure boundary and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually. This specification is applicable during Run, Startup, Shutdown, and Refuel modes of operation depending on the protective function. Table 3.1.1 Section A, is proposed for deletion in PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. There are no Modes of Operation once the reactor is defueled. There will no longer be a need for RPS since the postulated accident scenarios requiring actuation of RPS will no longer be possible. This TS does not provide protection for the cladding of fuel stored in the SFP; therefore, this TS may be deleted.
Table 3.1.1, Section B, Reactor Isolation	This system was designed to isolate the reactor by closing the main steam isolation valves and isolation condenser vent valves on indications of a pipe break in order to minimize coolant loss. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation depending on the protective function. Table 3.1.1, Section B, is proposed for deletion in PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The accident scenarios requiring actuation of the reactor isolation system will no longer be possible. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section C, Isolation Condenser Initiation	The isolation condenser system was designed to control reactor temperature and pressure when the reactor is isolated from its normal heat removal system. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section C, is proposed for deletion in PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Scenarios requiring actuation of the isolation condenser system will no longer be possible.

	There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section D, Core Spray	This system was designed to initiate the core spray system to provide cooling to the reactor in the event of the design basis LOCA. The system is designed to prevent or mitigate the consequences of postulated accidents that could result in damage to the reactor fuel or the reactor coolant pressure boundary. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section D, is proposed for deletion in PDTS. As discussed in 10 CFR 50.46(a)(1)(i), the requirement to have an Emergency Core Cooling System (ECCS) does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted. Therefore, the need for this associated protective instrumentation system is not required and this specification may be deleted. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section E, Containment Spray	The containment spray system was designed to control primary containment temperature and pressure during a LOCA. The automatic initiation requirement was revised to refer to compliance with TS 3.4 by License Amendment No. 160. Since TS 3.4 is also proposed for deletion (refer to TS 3.4 for bases for its deletion), Table 3.1.1, Section E, would be no longer meaningful, therefore it is also proposed for deletion in PDTS. This change is editorial in nature to align with the deletion of TS 3.4.
Table 3.1.1, Section F, Primary Containment Isolation	This specification provides the operability requirements for the primary containment isolation instrumentation, which automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section F, is proposed for deletion in PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The FHA in the SFP will be the only credible DBA possible in the permanently defueled condition. The FHA in the SFP analysis does not rely on primary containment to mitigate the consequences of the FHA. Therefore, this specification may be deleted. There are no Modes of Operation once the reactor is permanently defueled.
Table 3.1.1, Section G, Automatic Depressurization	This system was designed to initiate opening of the Electromatic Relief Valves to depressurize the reactor in the event of certain small break LOCA scenarios. This would allow the low pressure core spray system to operate for core cooling. The automatic depressurization system is designed to prevent or mitigate the consequences of postulated accidents

	that could result in damage to the reactor fuel. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section G, is proposed for deletion in PDTS. As discussed in 10 CFR 50.46(a)(1)(i), the requirement to have an ECCS does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted. Therefore, the need for an associated protective instrumentation system is not required and this specification may be deleted. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section H, Isolation Condenser Isolation	The isolation condenser isolation function was designed to sense a break in an isolation condenser pipe and close the isolation valves to isolate the break. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section H, is proposed for deletion in PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. A pipe break requiring actuation of the isolation condenser isolation system will no longer be possible. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section I, Offgas System Isolation	This system was designed to monitor offgas radiation levels during power operation and indicate when limits for the release of radioactive materials to the environment were approached. The offgas radiation monitoring system affects appropriate control of offgas flow so that the limits are not exceeded during plant operation. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section I, is proposed for deletion in PDTS. Since OCNGS will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the offgas monitors are no longer required. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section J, Reactor Building Isolation and Standby Gas Treatment System Initiation	These systems were designed to monitor radiation levels in the reactor building ventilation exhaust duct and in the vicinity of the fuel pool. These systems were also designed to initiate isolation signals to the reactor building ventilation and to initiate operation of the Standby Gas Treatment System (SGTS) when limits to the release of radioactive materials to the environment were approached. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section J, is proposed for deletion in PDTS. This specification will no longer be needed once all fuel has been moved to the SFP and 60-days have elapsed from the time of plant shutdown. In support of these changes, Exelon has completed an FHA in the SFP analysis (Reference 6) using the guidelines detailed in Regulatory Guide 1.183. The analysis

	demonstrates that radiological doses at the exclusion area boundary, low population zone and in the control room from a FHA after 60 days following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system. There are no Modes of Operation once the reactor is permanently defueled.
Table 3.1.1, Section K, Rod Block	This system was designed to maintain reactor thermal power within design limits by preventing the establishment of an abnormal rod pattern. This specification is applicable during Run, Startup, and Refuel Modes of operation. Table 3.1.1, Section K, is proposed for deletion in PDTS. Since OCNGS will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the control rod system is not required to control core reactivity and the control rod block actuation is no longer required. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section L, Condenser Vacuum Pump Isolation	Table 3.1.1, Section L was deleted by License Amendment No. 169 and serves as a placeholder. Placeholder specification titles are being proposed for deletion. This change is editorial in nature.
Table 3.1.1, Section M, Diesel Generator Load Sequence Timers	In the event of a loss of preferred power, safety system electrical loads are automatically connected to the diesel generators in sufficient time to provide for reactor safe shutdown and accident mitigation. Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading the diesel generators in the process. The settings of the sequence timers provide this function. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section M, is proposed for deletion in PDTS. As discussed in the basis for removal of TS 3.4 and 3.7, both ECCS and Emergency Diesel Generators (EDGs) will no longer be required once the plant is in the permanently defueled condition. Therefore, this specification will no longer be necessary and may be deleted. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section N, Loss of Power	This system was designed to assure that adequate power is available to operate emergency safeguards equipment. The loss of power instrumentation monitors 4160 volt safety division buses. The FHA in the SFP has been analyzed as the remaining design basis accident. Emergency safeguards equipment is not relied on to prevent the occurrence of or to mitigate the consequences of a FHA in the SFP. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section N. is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the

	certifications required by 10 CFR 50.82(a)(1) have been submitted, there is no need for loss of power instrumentation to ensure adequate power to emergency safeguards equipment. There are no Modes of Operation once the reactor is permanently defueled.
Table 3.1.1, Section O, Containment Vent and Purge Isolation	This specification establishes requirements for the primary containment vent and purge lines closure on high radiation in the drywell. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section O, is proposed for deletion in PDTS. Since OCNGS will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the source of high radiation in the drywell has been removed. Therefore, the drywell radiation monitor is no longer required. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Section P, RWCU HELB Isolation	This specification establishes requirements for Reactor Water Clean-Up (RWCU) isolation valve closure in the event of a high energy line break (HELB) in the RWCU system. Temperature switches at the entrance of the RWCU Pump Room are there to detect a line break downstream of the RWCU isolation valves. A line break will raise room temperature and before the room temperature exceeds 180°F, the switches will trip and close the RWCU isolation valves. System isolation will minimize the impact on off-site releases and the environmental qualification profiles for the Reactor Building. This specification is applicable during Run, Startup, Shutdown, and Refuel Modes of operation. Table 3.1.1, Section P, is proposed for deletion in PDTS. Since OCNGS will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the RWCU system will no longer be required or contain high temperature radioactive fluid. There are no Modes of Operation once the reactor is permanently defueled. This TS does not provide protection for the fuel stored in the SFP.
Table 3.1.1, Notes	This specification contains the "Notes" that are applicable to Table 3.1.1. They provide additional clarification to the requirements given in Table 3.1.1. Since all the conditions in Table 3.3.1 are proposed for deletion in the PDTS, the "Notes" section for Table 3.1.1 is also being proposed for deletion. This change is administrative in that the “Notes” will serve no meaningful function once the condition to which they are applicable has been deleted.

Proposed OCNGS TS Section 3/4.1 SPENT FUEL STORAGE

Applicability: ***During movement of irradiated fuel assemblies in the spent fuel pool.***

Objective: ***To assure safe storage of spent fuel.***

LCO: 3.1 ***Spent Fuel Pool Water Level***

Whenever irradiated fuel is stored in the spent fuel storage pool, water level shall be maintained at a level ≥ 117 feet 8 inches (elevation above sea level) with the exception of planned cask movements.

ACTIONS:

Condition	***Required Action***	***Completion Time***
Spent fuel pool water level is not within limit.	***Suspend movement of irradiated fuel assemblies and movement of loads over the storage racks containing fuel.***	***Immediately***

SURVEILLANCE REQUIREMENTS

Surveillance		***Frequency***
4.1	***Verify the spent fuel pool water level is ≥ 117 feet 8 inches.***	***24 hours***

Basis

A specification for SFP water level is being proposed to ensure safe storage and management of the spent fuel. This specification is being numbered as TS 3/4.1. The table of contents is also revised to reflect these changes.

LCO 3.1, "Spent Fuel Pool Water Level," specifies requirements to ensure that the minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident (FHA) in the spent fuel pool (SFP). The water also provides shielding during the movement of spent fuel.

The required minimum water level in the SFP meets the assumptions of the FHA described in calculation C-1302-226-E310-460 and Chapter 15.7.4 of the UFSAR. The resultant dose limits at the exclusion area boundary are within the criteria of RG1.183.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2. The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.4.

The FHA is evaluated for dropping an irradiated fuel assembly onto irradiated fuel bundles stored in the SFP. The consequences of a FHA in the SFP are documented in FSAR Chapter 15. The water level in the SFP provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the building atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a FHA.

The SFP water level is monitored in terms of elevation above mean sea level. Elevation 117 feet 8 inches corresponds to the SFP low level alarm in the Control Room. Since the pool has no installed drains, level cannot be lowered by the cooling system below the level of the weirs. At the normal 400 gpm flow rate, the pool level is about three inches above the weir level, and the overflow just equals the 400 gpm being supplied to the pool from the diffusers. At the SPF low level alarm level, the pool contains a depth of approximately 37 feet of water (approximately 23 feet above active fuel), providing adequate shielding for normal building occupancy by operating personnel.

LCO 3.1 requires that when the water level in the SFP is lower than the required level, the movement of irradiated fuel assemblies in the SFP is to be "immediately" suspended. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, such that the suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring in the SFP when the level is below the required elevation.

This specification is not meant to affect spent fuel cask movements during planned SFP level adjustments. The FSAR Chapter 15 analysis states that a spent fuel cask drop accident is no longer credible since the reactor building crane has been upgraded to be single-failure proof.

Surveillance Requirement (SR) 4.1 verifies that sufficient SFP water is available in the event of a fuel handling accident. The water level in the SFP must be checked periodically. The frequency of every 24 hours is acceptable based on operating experience, considering that the water volume in the pool is normally stable and water level changes are controlled by unit procedures.

The fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS SECTION 3.2 – REACTIVITY CONTROL

TS Section 3.2 "Reactivity Control," contains LCOs related to reactivity control capability and applies to core reactivity and the reactivity control systems to protect the integrity of the fission product barrier. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the reactivity control functions addressed in TS Section 3.2 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
TS 3.2.A, Core Reactivity	This specification defines the minimum shutdown margin (SDM) in the reactor core during all modes of operation. Core Reactivity ensures that the core loading is limited to that which can be made subcritical in the most reactive condition during the operation cycle, with the highest worth operable control rod in its fully withdrawn position and all other operable rods inserted. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.2.A is proposed for deletion in PDTS. Since OCNGS will no longer be authorized operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactivity limitations

	associated with fuel in the reactor will not apply. This TS applies only when fuel is in the reactor vessel core.
TS 3.2.B, Control Rod System	This specification defines the operability requirements for the control rods and establishes conditions to prevent the addition of excess reactivity due improper operation of control rods or failure of the control rod system. This specification is applicable during power operation and when the RCS is pressurized above atmospheric pressure with fuel in the reactor. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.2.B is proposed for deletion in PDTS. Since OCNGS will no longer be authorized operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), control of reactivity in the reactor core will no longer be relevant. Therefore, there is no need to place conditions for operation on the control rod system.
TS 3.2.C, Standby Liquid Control System	This specification defines the operability requirements for the Standby Liquid Control (SLC) system during all modes of operation. It ensures that: 1) the SLC system has the capability to independently shutdown the reactor from full power to a cold, xenon-free shutdown assuming none of the withdrawn control rods could be inserted; and 2) the SLC system meets the requirement of the Anticipated Transient Without Scram (ATWS) rule (10 CFR 50.62). This specification is applicable when the reactor is not shutdown with control rods and is > 212°F. This specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.2.C, including Figures 3.2-1, "Sodium Pentaborate Solution Volume-Concentration Requirement" and Figure 3.2-2, "Sodium Pentaborate Solution Temperature Requirements," is proposed for deletion in PDTS. Since OCNGS will no longer be authorized operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the requirements of 10 CFR 50.62 (ATWS) will no longer apply to OCNGS. There is no need for a standby liquid control system.
TS 3.2.D, Reactivity Anomalies	This specification establishes conditions to limit the difference in observed and predicted control rod inventory to no greater than one percent equivalent reactivity. This limit ensures that the reactivity inserted into the core, in the worst case, would not lead to transients exceeding the design conditions of the reactor system. This specification is applicable when the reactor is not shutdown and is > 212°F. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.2.D is proposed for deletion in PDTS. Since OCNGS will no longer be authorized operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactivity anomalies are no longer be applicable.

TS SECTION 3.3 – REACTOR COOLANT

TS Section 3.3 "Reactivity Coolant," contains LCOs that provide assurance of the reactor coolant pressure boundary (RCPB) integrity and safe operation of the reactor coolant system (RCS). The protection and monitoring functions of the RCS has been designed to ensure safe operation of the reactor required to protect the integrity of a fission product barrier. The RCS is a primary barrier against the release of fission products to the environs.

<table>
<tr><td colspan="2">10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," stipulates that reactor facilities which have submitted the certifications required under § 50.82(a)(1), no longer need to meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H. The maintenance rule (10 CFR 50.65) will be used to monitor the performance or condition of the SSCs associated with the storage, control, and maintenance of spent fuel in a safe condition. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the reactor coolant specifications addressed in TS Section 3.3 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.</td></tr>
<tr><th>Current OCNGS TS</th><th>Basis for Deletion</th></tr>
<tr><td>TS 3.3.A, Pressure Temperature Relationships</td><td>This specification establishes the limits associated with maintaining the vessel pressure and temperature limits including the limitation established with heatup and cooldown rates to prevent encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB. This limitation is applicable at all times. The RCS heatup and cooldown rate limits, contained in the Pressure and Temperature Limits Report (PTLR) described in TS 6.23, provide a definition of acceptable operation for prevention of nonductile failure pursuant to 10 CFR Part 50, Appendix G. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS 3.3.A and the PTLR discussed in TS 6.23 are proposed for deletion in PDTS. The requirements of 10 CFR Part 50, Appendix G no longer apply because the RCPB will no longer be used as a fission product barrier when the reactor vessel is permanently defueled. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these limitations are no longer needed to protect the reactor vessel and preserve the integrity of the RCPB.</td></tr>
<tr><td>TS 3.3.B, Reactor Vessel Closure Head Boltdown</td><td>This specification places limits on reactor vessel head stud tensioning to ensure analyzed stress and fatigue values are not exceeded. The objective of this specification is to assure the structure integrity of the RCS. Stud tensioning parameters are specified in the PTLR described in TS 6.23. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS 3.3.B and the PTLR discussed in TS 6.23 is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these limits are no longer needed to protect the reactor vessel and preserve the integrity of the reactor coolant pressure boundary. Therefore, there is no need to establish reactor vessel stud tensioning requirements.</td></tr>
<tr><td>TS 3.3.C, Thermal Transients</td><td>This specification provides limits on the rate of temperature change during heatup and cooldown of the reactor vessel and the temperature of coolant in an idle recirculation loop to be started. The objective of this specification is to assure the structural integrity of the RCS. The limit on the rate of temperature change during heatups and cooldowns ensures that the</td></tr>
</table>

	assumption of the reactor vessel stress analysis is valid. The temperature limit on coolant in an idle recirculation loop ensures that the cold water addition transient remains within analyzed bounds. The average rate of reactor coolant change values are specified in the PTLR described in TS 6.23. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.3.C and the PTLR discussed in TS 6.23 is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these limits are no longer needed to protect the reactor vessel and limit a reactivity addition.
TS 3.3.D, Reactor Coolant System Leakage	This specification establishes process variable limits and operating restrictions for unidentified and identified RCS leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a DBA. This TS is applicable when the RCS temperature is > 212°F. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.3.D is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), limits on primary system leakage is no longer of concern.
TS 3.3.E, Reactor Coolant Quality	This specification establishes requirements for reactor coolant quality. Reactor coolant system water chemistry is monitored and limits established at different steaming rates for conductivity and chlorides to prevent stress corrosion cracking of 304 stainless steel materials and damage to Zircaloy fuel cladding. If those limits are not met, an orderly shutdown is initiated. This TS is applicable when the RCS temperature is > 212°F. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.3.E is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), there is no need to monitor reactor coolant system chemistry.
TS 3.3.F, Recirculation Loop Operability	This specification establishes requirements for the availability of recirculation loops to ensure reactor core flow remains within analyzed limits during power operation and to ensure the operability of reactor water level monitoring instruments that are used for reactor protection. The TS is applicable during Power Operation of the plant. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.3.F is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), there is no need to establish requirements on reactor recirculation loops.
TS 3.3.G, Primary Coolant System Pressure Isolation Valves	This specification establishes leakage limits on certain reactor coolant system isolation valves, listed on Table 3.3.1, to increase the reliability of these valves in reducing the potential of an inter-system loss of coolant accident during Power Operation. This specification is applicable in the Startup and Run Modes. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

	TS 3.3.G, including Table 3.3.1, is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), there is no need to establish requirements on reactor pressure isolation valves.
TS 3.3.H, Required Minimum Recirculation Flow Rate for Operation in IRM Range 10	This specification requires a minimum reactor recirculation flow during power operation in the Startup Mode when entering IRM range 10. This ensures that transient Minimum Critical Power Ratio (MCPR) limits for reactor operation are not exceeded. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.3.H is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), there is no need to establish reactor recirculation flow requirement for IRM range 10.

TS SECTION 3.4 – EMERGENCY COOLING

TS Section 3.4 "Emergency Cooling," contains LCOs to assure the operability of the emergency cooling systems and to provide assurance of adequate cooling capability for heat removal in the event of a LOCA or isolation from the normal reactor heat sink. The table below describes the specifications in this section.

10 CFR 50.46, *"Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,"* specifies that light-water nuclear power reactors must have ECCS designed to meet the cooling performance requirements following postulated LOCAs. 10 CFR 50.46(a)(1)(i) states "This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted."

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the core and containment cooling specifications addressed in TS Section 3.4 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.4.A, Core Spray System	This specification establishes requirements for the Core Spray System (CSS). The CSS is part of the emergency core cooling systems and is designed to provide sufficient cooling to the core to dissipate decay heat after a LOCA and limit fuel clad temperature and oxidation. Core geometry remains intact and fuel clad metal-water reaction is less than 1%. The CSS is required to operable at all times with fuel in the reactor vessel with an absorption chamber water level of at least 82,000 ft^3 except as specified in Table 3.4.1. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.4.A, including Table 3.4.1, is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), a LOCA is no longer possible and emergency core cooling systems are no longer needed.
3.4.B, Automatic Depressurization System	This specification establishes requirements for the Automatic Depressurization System (ADS). ADS is a subsystem that complements the

	core spray system and provides protection against small pipe breaks by depressurizing the reactor vessel rapidly to actuate core spray. ADS is required to operable when reactor water temperature is > 212 F and pressure is above 110 psig. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.4.B is proposed for deletion in PDTS. Since OCNGS has permanently ceased power operations and is no longer authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the postulated small pipe break accidents that would require initiation of the ADS subsystem are no longer possible.
3.4.C, Containment Spray System and Emergency Service Water System	This specification establishes requirements for the Containment Spray and Emergency Service Water System. These systems are designed to remove heat energy from primary containment in the event of a LOCA. The Containment Spray and Emergency Service Water Systems are required to be operable at all times with fuel in the reactor vessel and an absorption chamber water level of at least 82,000 ft^3. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.4.C. is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), a LOCA that requires initiation of containment cooling systems is no longer possible.
3.4.D, Control Rod Drive Hydraulic System	This specification establishes requirements for Control Rod Drive Hydraulic System. The control rod drive pumps supply hydraulic pressure to the control rod drive system to position control rods for reactivity adjustments in the reactor core. The reactor coolant provided to the control rod drives is exhausted to the reactor vessel and provides the capability for a small amount of makeup in the event of very small pipe breaks. Accident analysis does not take credit for the makeup provided by the control rod drive pumps. The Control Rod Drive Hydraulic system is required to operable when reactor water temperature is greater than 212 F. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.4.D is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), control rod adjustments will not be required and a LOCA is no longer possible. Therefore, the control rod drive hydraulic system is no longer needed.
3.4.E, Core Spray and Containment Spray Pump Compartments Doors	This specification establishes requirements for Core Spray and Containment Spray Pump Compartment Doors. This specification ensures that the core spray and containment spray pump compartment doors are closed in order to consider the respective systems operable. The doors prevent flooding from the torus room should the torus develop a leak. The compartment doors are required to be closed at all times. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.4.E is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), a LOCA is no longer possible and the core spray and containment spray systems are not necessary. Therefore, flooding protection of these systems and assured closure of the pump compartment doors are longer needed.

3.4.F, Fire Protection System	This specification assures the fire protection system operability. The fire protection system incorporates a tie-in to the core spray system to provide a water supply backup for core cooling in the unlikely event the core spray system pumps are unavailable. The fire protection system has limited capability as a backup core cooling system and does not meet regulatory requirements for an Emergency Core Cooling System. TS 3.4.F is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), a LOCA is no longer possible and emergency core cooling systems are no longer needed. Therefore, the fire protection system backup is no longer needed.

TS SECTION 3.5 – CONTAINMENT

TS Section 3.5 "Containment," contains LCOs that assure the integrity of the primary and secondary containment systems. The primary containment system provides a barrier against uncontrolled release of fission products to the environs in the event of a LOCA. Secondary containment is designed to minimize any ground level release of radioactive materials that might result from an accident.

The analyzed DBA that remains applicable to OCNGS in the permanently shut down and defueled condition is a FHA in the SFP. A calculation (C-1302-226-E310-460, "EAB, LPZ, and CR Dose Due to Fuel Handling Accident (FHA) - Post Cessation of Power Operations" (Reference 6)) was performed to assess the dose consequences of a postulated FHA after cessation of power operations. The calculation demonstrates that radiological doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the Control Room (CR) are within allowable limits of 10 CFR 50.67 without crediting secondary containment operability, standby gas treatment system, or CR high efficiency air filtration after a 60-day fuel decay period following permanent reactor shutdown.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications for the primary and secondary containment systems addressed in TS Section 3.5 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	**Basis for Deletion**
3.5.A, Primary Containment	This specification establishes primary containment integrity when the reactor is critical or when the reactor is at power or hot shutdown. These limits ensured that primary containment integrity is maintained to prevent the release of fission products to the environment in the event of a LOCA. The primary containment is required to be operable during any operational condition where the temperature is greater than 212°F, or when work is performed with fuel in the reactor that has the potential to drain the vessel. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Additionally, operability requirements for Reactor Building to Suppression Chamber Vacuum Breakers, Suppression Chamber to Drywell Vacuum Breakers and Shock Suppressors (Snubbers) are also included in this section. The Vacuum Breaker Systems support maintaining the integrity of the Primary Containment. Snubbers were designed to limit movement and prevent damage to reactor coolant system components during all modes of reactor operation, including heatups and cooldowns. All safety-related

	snubbers are required to be operable whenever the systems they protect are operable. Since OCNGS will be permanently shut down and defueled, these support systems are no longer required since maintaining primary containment integrity is no longer required. TS 3.5.A is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the potential for a release of fission products to the environment in the event of a LOCA no longer exist. Therefore, maintaining primary containment integrity is no longer applicable.
3.5.B, Secondary Containment	This specification establishes the requirements for secondary containment integrity. Secondary containment is designed to minimize any ground level release of radioactive materials that might result from an accident. Secondary containment integrity is required whenever primary containment integrity is required and whenever activities having the potential of significant fission product release, such as moving irradiated fuel or performing activities in the reactor building or around the SFP that could cause the release of radioactive materials. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). This section also contains requirements for the standby gas treatment system (SGTS) that is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. The SBGT system shall be operable when secondary containment is required. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Accident analysis for the permanently defueled condition is described in Chapter 15 of the UFSAR. The FHA in the SFP was analyzed and the radiological consequences were bounded by fuel failure caused by a dropped bundle. The radiological consequences in the CR are within allowable limits of 10 CFR 50.67 without crediting secondary containment operability, standby gas treatment system, or CR high efficiency air filtration after a 60-day fuel decay period following permanent reactor shutdown. TS 3.5.B is proposed for deletion in PDTS. Since OCNGS accident analysis no longer relies on secondary containment integrity and the SGTS for mitigation of radiological releases, these systems are no longer required.

TS SECTION 3.6 – RADIOACTIVE EFFLUENTS

TS Section 3.6 "Radioactive Effluents," contains LCOs to assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that the radioactive concentrations of any material released is kept as low as reasonably achievable and, in any event, within the limits of 10 CFR 20.1301 and 40 CFR Part 190.10(a). The LCOs in this section apply to radioactive effluents of the facility. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety, except for specification 3.6.C which remains applicable and is being renumbered as PDTS 3.2, as indicated in the Basis for Deletion column below. Many of the specifications in TS 3.6 have been previously relocated to the ODCM as discussed below. The placeholders associated with these specifications are being removed in the proposed PDTS; this proposed change is editorial. The remainder of the specifications in TS Section 3.6 do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, as discussed in the table

below, the remainder of the radioactive effluents TS addressed in TS Section 3.6 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.6.A, Reactor Coolant Radioactivity	This specification provides limits regarding the specific activity monitored in the reactor coolant in all reactor modes except refuel. Dose Equivalent I-131 is used to express dose from a mixture of iodine isotopes created in an operating core and contained in reactor coolant. This is done to ensure radioactivity in reactor coolant that may be released during postulated events results in offsite doses that are acceptably low and to identify fuel leaks and failures. This specification is required when the reactor is critical. These limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.6.A is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), there is no need to monitor reactor coolant activity. The accident conditions will no longer apply and monitoring the value of Dose Equivalent I-131 for dose analysis of accidents involving primary coolant releases will no longer be required.
3.6.B, Liquid Radwaste Treatment	This specification was relocated to the Offsite Dose Calculation Manual (ODCM) in TS Amendment 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.C, Radioactive Liquid Storage	TS Section 3.6.C - Radioactive Liquid Storage - This specification limits the amount of radioactive material contained in storage tanks that could leak to the environment. This specification and the corresponding SR are being modified and renumbered as TS 3/4.2 in the PDTS. See discussion for TS 3/4.2 below.
3.6.D, Condenser Offgas Treatment	This specification was relocated to the ODCM in TS Amendment No. 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.E, Main Condenser Offgas Radioactivity	This specification provides the operability requirements for the main condenser offgas radioactivity. Condenser offgas is a byproduct of the nuclear reaction in the reactor core during power operation. This specification provides a limit on gross radioactivity in the noble gases. This specification is required when the reactor is critical. These limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS Section 3.6.E is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), a vacuum in the main condenser will not be established and the condenser offgas will no longer be generated. Therefore, this specification is not required.
3.6.F, Condenser Offgas Hydrogen Concentration	This specification provides the limits and monitoring requirements for hydrogen concentration in the offgas downstream of the operating recombiner in the augmented offgas system. The hydrogen monitors are

	used to detect possible hydrogen buildups, which could result in a possible hydrogen explosion. This TS applies in all modes of normal operation. TS Section 3.6.F is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), hydrogen will no longer be produced by reactor operation, a vacuum in the main condenser will not be established, and the condenser offgas will no longer be generated. Therefore, this specification is not required.
3.6.G, Not Used	Specification numbers 3.6.G is not used, therefore, is proposed for deletion. This change is editorial in nature.
3.6.H, Not Used	Specification numbers 3.6.H is not used, therefore, is proposed for deletion. This change is editorial in nature.
3.6.I, Radioactivity Concentration in Liquid Effluent	This specification was relocated to the ODCM in TS Amendment 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.J, Limit on Dose Due to Liquid Effluent	This specification was relocated to the ODCM in TS Amendment No. 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.K, Dose Rate Due to Gaseous Effluent	This specification was relocated to the ODCM in TS Amendment No. 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.L, Air Dose Due to Noble Gas in Gaseous Effluent	This specification was relocated to the ODCM in TS Amendment No. 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.M, Dose Due to Radioiodine and Particulates in Gaseous Effluent	This specification was relocated to the ODCM in TS Amendment No. 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, placeholder specification titles are proposed for deletion. This change is editorial in nature.
3.6.N, Annual Total Dose Due to Radioactive Effluents	This specification was relocated to the ODCM in TS Amendment No. 166, dated December 13, 1993 (ADAMS Accession No. ML011200256). Therefore, these placeholder specification titles are proposed for deletion. This change is editorial in nature.

Proposed OCNGS TS 3/4.2 RADIOACTIVE LIQUID STORAGE

Applicability: Applies at all times to ~~specified~~ outdoor tanks used to store radioactive liquids.

Objective: To assure that radioactive ~~material is~~ ***effluents are*** not released to the environment in an uncontrolled manner and to assure that the radioactive concentrations of any material released is kept as low as is reasonably achievable and, in any event, within the limits of 10 CFR ***P***art 20.1301 and 40 CFR Part 190.10(a).

LCO: 3.2 The quantity of radioactive material, excluding tritium, noble gases, and radionuclides having half-lives shorter than three days, contained in ~~any of the following~~ outdoor ***storage*** tanks shall not exceed 10.0 curies~~:~~***. Included in this specification are all outdoor storage tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.***

~~a. Waste Surge Tank, HP-T-3~~

~~b. Condensate Storage Tank~~

ACTIONS:

Condition	***Required Action***	***Completion Time***
In the event the quantity of radioactive material in any ~~of the~~ ***applicable storage tanks*** ~~named~~ ***exceeds 10.0 curies.***	Begin treatment and continue it until the total quantity of radioactive material in the tank is 10 curies or less, and describe the reason for exceeding the limit in the next Annual Effluent Release Report.	As soon as reasonably achievable

SURVEILLANCE REQUIREMENTS

Surveillance		***Frequency***
4.2	Liquids contained in ~~the following~~ ***outdoor storage*** tanks ***included in this specification*** 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.~~	***Once per 7 days when radioactive liquid is being added to the tank.***

Basis
The current TS 3.6.C for Radioactive Liquid Storage is being modified and renumbered as proposed TS 3/4.2 to ensure safe storage and management of radioactive liquids contained in outdoor storage tanks. The table of contents is also revised to reflect these changes. The two specified tanks are being deleted from the TS in order to broaden the definition of tanks. The TS would include all outdoor storage tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system. The associated Surveillance Requirement is relocated as part of the reformatting of the TS to combine Sections 3 and 4. The Surveillance Frequency is the frequency specified in the Surveillance Frequency Control Program, which is being deleted for PDTS (see TS 6.24 below). The specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

TS SECTION 3.7 – AUXILARY ELECTRIC POWER

TS Section 3.7 "Auxiliary Electric Power," contains LCOs related to the operability of AC and DC electrical systems. The Section establishes the requirements for appropriate functional capability of plant electrical equipment required for safe operation of the facility. This section specifies requirements to ensure that the station safety-related electrical bussing and distribution system, offsite power sources, and the onsite standby power sources (emergency diesel generators (EDG)), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to engineered safety features (ESF) systems so that the fuel, RCS, and containment design limits are not exceeded. The requirements for the EDG fuel oil storage are included for each EDG. Also included in this section is the requirements for DC power. It specifies requirements to ensure that the DC electrical power subsystems are operable. The table below describes the specifications in this section.

The design basis accidents and transients analyzed in UFSAR Chapter 15 will no longer be applicable in the permanently defueled condition, with the exception of the FHA in the SFP. Exelon performed a calculation (Reference 6) for a FHA in the SFP that shows the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event (after 60 days of irradiated fuel decay time after reactor shutdown and compliance with the SFP water level requirements in proposed TS 3/4.1).

Because the FHA analysis does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. Therefore, during movement of irradiated fuel assemblies in the SFP, there are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA with the unit permanently defueled. As such, the requirement for AC and DC sources are being deleted because there are no design basis events that rely on these sources for mitigation.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.7 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.7.A, Required Electrical Sources	This specification provides the AC and DC electrical power requirements during Startup/Hot Standby and when the reactor is operating. These limits

	ensure that the offsite power sources, the onsite standby power sources, emergency buses and RPS power protection provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to engineered safety features systems during and following DBAs so that the fuel, RCS, and containment design limits are not exceeded. These specifications satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the NRC that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this condition, the operational conditions, transients, and postulated DBAs are no longer possible. Therefore, the systems required for reactor safety, which the auxiliary electrical systems were designed to power, are no longer needed. The only DBA that could currently apply to the permanently shutdown and defueled OCNGS reactor would be the FHA. AC and DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA in the SFP. Because the FHA analysis does not rely on AC or DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), these sources are therefore not required. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA with OCNGS permanently shutdown and defueled. The only electrically powered active system important for the storage of irradiated fuel is the SFP cooling and support systems. The SPF cooling system did not meet the criteria in 10 CFR 50.36 for inclusion in the OCNGS TS even when the reactor was authorized to operate. Thus, this specification is not being proposed for inclusion in the PDTS since the DBAs that require power for emergency safeguards systems supplied by the auxiliary electrical system are no longer applicable in the permanently defueled condition.
3.7.B, Required Actions for 3.7.A	This specification provides requirements for continued operation of the reactor when the availability of power falls below that required in TS 3.7.A. As stated above (TS 3.7.A), AC and DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA in the SFP. Thus, this specification is not being proposed for inclusion in the PDTS since the DBAs that require power for emergency safeguards systems supplied by the auxiliary electrical system are no longer applicable in the permanently defueled condition.
3.7.C, Standby Diesel Generators	TS Section 3.7.C - Standby Diesel Generators - This LCO specifies diesel generator operability requirement in support of reactor operation. This specification is applicable in all modes of operation. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the NRC that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this condition, the operational conditions, transients, and postulated DBAs are

	no longer possible. Therefore, the systems required for reactor safety that the auxiliary electrical power systems were designed to power are no longer needed. The only DBA that could currently apply to the permanently shutdown and defueled OCNGS reactor would be the FHA. AC and DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA in the SFP. Because the FHA analysis does not rely on AC or DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), these sources are therefore not required. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA with OCNGS permanently shutdown and defueled. The only electrically powered active system important for the storage of irradiated fuel is the SFP cooling and support systems. The SPF cooling system did not meet the criteria in 10 CFR 50.36 for inclusion in the OCNGS TS even when the reactor was authorized to operate. Therefore, there is no need to maintain emergency AC power requirements.
3.7.D, Station Batteries and Associated Battery Chargers	This specification provides operability requirements for the safety-related batteries (B and C) and battery chargers. The LCOs provide required actions for maintaining battery terminal voltage greater than or equal to the minimum float change, cell electrolyte level, and electrolyte temperature. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the NRC that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this condition, the operational conditions, transients, and postulated DBAs are no longer possible. Therefore, the systems required for reactor safety that the auxiliary electrical power systems were designed to power are no longer needed. The only DBA that could currently apply to the permanently shutdown and defueled OCNGS reactor would be the FHA. AC and DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA in the SFP. Because the FHA analysis does not rely on AC or DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), these sources are therefore not required. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHAwith OCNGS permanently shutdown and defueled. The only electrically powered active system important for the storage of irradiated fuel is the SFP cooling and support systems. The SPF cooling system did not meet the criteria in 10 CFR 50.36 for inclusion in the OCNGS TS even when the reactor was authorized to operate. Therefore, there is no need to maintain emergency DC power requirements.

TS SECTION 3.8 – ISOLATION CONDENSER

TS Section 3.8 "Isolation Condenser," contains LCOs related to the operability requirements for the isolation condenser and their isolation valves. The isolation condenser assures decay heat removal from the reactor core under conditions when the reactor vessel is isolated from its normal heat sink. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.8 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.	
Current OCNGS TS	**Basis for Deletion**
3.8.A, Two Isolation Condenser Loops	This specification requires two isolation condenser loops to be operable during power operations and whenever the reactor coolant temperature is greater than 212°F. TS 3.8.A is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain a safe shutdown and remove decay heat no longer exists. Therefore, there is no need for the isolation condensers and these specifications are proposed for deletion.
3.8.B, Minimum Water Volume – Condenser Shell Side	This specification requires the shell side of each condenser to contain a minimum water volume of 22,730 gallons. If the minimum volume cannot be maintained or if a source of makeup water is not available to the condenser, the condenser shall be considered inoperable. TS 3.8.B is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain a safe shutdown and remove decay heat no longer exists. Therefore, there is no need for the isolation condensers and these specifications are proposed for deletion.
3.8.C, With One Isolation Condenser Inoperable in Run Mode	This specification allows one isolation condenser to become inoperable during the run mode if certain conditions are met. TS 3.8.C is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain a safe shutdown and remove decay heat no longer exists. Therefore, there is no need for the isolation condensers and these specifications are proposed for deletion.
3.8.D, Required Action if Specification 3.8.A and 3.8.B not met	This specification requires the reactor to be shutdown and placed in a cold shutdown condition if certain conditions cannot be met. TS 3.8.D is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain a safe shutdown and remove decay heat no longer exists. Therefore, there is no need for the isolation condensers and these specifications are proposed for deletion.
3.8.E, Inoperable Isolation Condenser Inlet (Steam Side) Isolation Valve	This specification allows reduction in redundancy of isolation capability for isolation condenser inlet (steam side) isolation valves. TS 3.8.E is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to

	operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain a safe shutdown and remove decay heat no longer exists. Therefore, there is no need for the isolation condensers and these specifications are proposed for deletion.
3.8.F, Inoperable AC Motor-Operated Isolation Condenser Outlet (Condensate Return) Isolation Valve	This specification allows short term inoperability of the AC motor-operated isolation condenser outlet (condensate return) valve. TS 3.8.F is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain a safe shutdown and remove decay heat no longer exists. Therefore, there is no need for the isolation condensers and these specifications are proposed for deletion.

TS SECTION 3.9 – REFUELING

TS Section 3.9 "Refueling," contains LCOs related to fuel handling operations during refueling. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur. The table below describes the specifications in this section.

The specifications of this section are being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.9 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
TS 3.9.A, Control Rod Position	This specification requires the control rod in a reactor core cell to be fully inserted when loading fuel assemblies into the core cell to minimize the probability of an inadvertent criticality during refueling. This specification is applicable during core alterations. The control rod position limit was included to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.A is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), core alterations will no longer occur or be permitted.
TS 3.9.B, Reactor Mode Switch	This specification requires the reactor mode switch to be locked in the Refuel position during core alterations. The mode switch in the refuel position allows one control rod only to be withdrawn. The control rod position limit was included to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.B is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), core alterations will no longer occur or be permitted.
TS 3.9.C, Refueling Equipment Interlocks	This specification provides the operability requirements for the refueling equipment interlocks. This specification is applicable during in-vessel fuel movement. The function of these interlocks is to restrict the operation of the frame-mounted auxiliary hoist, the trolley-mounted auxiliary hoist or the

	service platform hoist. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.C is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), requirements related to core alterations are no longer required.
TS 3.9.D, Source Range Monitors	This specification required two operable source range monitor (SRM) channels during core alterations. The source range monitors provide neutron flux monitoring capabilities with the reactor in the refueling and shutdown modes. The redundant monitoring capability is available to detect changes in the reactivity condition of the core. TS 3.9.D is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), requirements for core monitoring will no longer be required.
TS 3.9.E, Removal of Single Control Rod	This specification allows for the removal of one control rod or control rod drive mechanism provided certain requirements are met. The specification is applicable during Refueling. No criteria of 10 CFR 50.36(c)(2)(ii) apply. TS 3.9.E is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), requirements related to core alterations are no longer required.
TS 3.9.F, Removal of Multiple Control Rods	This specification allows for the removal of multiple control rods or control rod drive mechanisms provided certain requirements are met. The specification is applicable during Refueling. No criteria of 10 CFR 50.36(c)(2)(ii) apply. TS 3.9.F is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), requirements related to core alterations are no longer required.
TS 3.9.G, Any Refueling Requirement Not Met	This specification required that core alterations or control rod removal cease as appropriate, and actions be initiated to satisfy the above requirements. TS 3.9.G is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), requirements related to core alterations are no longer required.

TS SECTION 3.10 – CORE LIMITS

TS Section 3.10 "Core Limits," contains the LCOs to ensure that power distribution limits are met. The LCOs will not apply to a reactor that is in a permanently defueled condition. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.10 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.10.A, Average Planar Linear Heat Generation Rate (APLHGR)	TS Section 3.10.A defines limits for the APLHGR to ensure that the peak cladding temperature (PCT) during the postulated design basis Loss of Coolant Accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. This specification is applicable during Power Operation. The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.10.A is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these limits are no longer applicable. This TS does not provide protection for the cladding of fuel stored in the SFP.
3.10.B, Local Linear Heat Generation Rate (LHGR)	TS Section 3.10.B defines limits for the LHGR to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients, and to ensure that the PCT during a postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. This specification is applicable during Power Operation. The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.10.B is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these limits are no longer applicable. This TS does not provide protection for the cladding of fuel stored in the SFP.
3.10.C, Minimum Critical Power Ratio (MCPR)	TS 3.10.C defines limits for the MCPR to ensure that no fuel damage results during abnormal operational transients. This specification is applicable during Power Operation. The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.10.C is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these limits are no longer applicable. This TS does not provide protection for the cladding of fuel stored in the SFP.

TS SECTION 3.11

TS Section 3.11 was removed in License Amendment No. 29. The remaining reference to TS 3.11 is proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 3.12 – ALTERNATE SHUTDOWN MONITORING INSTRUMENTATION

TS Section 3.12 "Alternate Shutdown Monitoring Instrumentation," contains the LCOs related to alternate shutdown monitoring instrumentation from outside the main control room. TS Table 3.12-1 lists the required instrumentation in this section. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.12 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.12.A and 3.12.B and Table 3.12-1	The instrumentation identified in this specification ensure sufficient capability is available to permit shutdown and maintenance of hot shutdown of the plant from locations outside the control room. The specifications apply only when the plant is at power operation and when reactor coolant temperature is above 212°F. TS 3.12, including Table 3.12-1, is proposed for deletion in the PDTS since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The need to achieve and maintain hot shutdown are no longer applicable.

TS SECTION 3.13 – ACCIDENT MONITORING INSTRUMENTATION

TS Section 3.13 "Accident Monitoring Instrumentation," contains LCOs related to the operability during power operation or when primary containment integrity is required to monitor the course of reactor accidents. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.13, including Table 3.13.1, will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.13.A, Relief Valve Position Indicators	This specification requires the operability of the relief valve accident monitoring instrumentation in order to alert the operator to a stuck open relief valve which could result in an inventory threatening event. The accident monitoring instrumentation channels shown in Table 3.13.1 shall be Operable when the mode switch is in the Startup or Run positions. The specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.13.A, including Table 3.13.1, is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), relief valve position monitoring instrumentation channels are no longer applicable.

3.13.B, Safety Valve Position Indicators	This specification requires the operability of safety valve accident monitoring instrumentation in order to alert the operator to a stuck open safety valve which could result in an inventory threatening event. The primary and backup safety valve monitoring instruments shall be Operable during Power Operation. The specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.13.B is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), safety valve position indicators are no longer applicable.
3.13.C, Required Action for 3.13.A and 3.13.B	In the event that any of the safety/relief valve monitoring channels becomes inoperable, they shall be made Operable prior to startup following the next Cold Shutdown. TS 3.13.C is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), relief valve or safety valve position monitoring instruments are no longer applicable.
3.13.D, Wide Range Torus Water Level Monitor	This specification requires the operability of the accident monitoring instrumentation to ensure that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. Two wide range torus water level monitor channels listed in Table 3.13.1 shall be continuously indicated in the control room during Power Operation. TS 3.13.D, including Table 3.13.1, is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactor accidents and primary containment integrity are no longer applicable.
3.13.E, Wide Range Drywell Pressure Monitor	This specification requires the operability of the accident monitoring instrumentation to ensure that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. Two Wide Range Drywell Pressure monitor channels listed in Table 3.13.1 shall be continuously indicated in the control room during Power Operation. TS 3.13.E, including Table 3.13.1, is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactor accidents and primary containment integrity are no longer applicable.
3.13.F, Deleted	TS Section 3.13.F was removed in a previous license amendment. The remaining reference to TS 3.13.F is proposed to be removed as part of the editorial cleanup of the PDTS.
3.13.G, Containment High-Range Radiation Monitor	This specification requires containment post-accident monitoring capability through two high range radiation monitors installed in the drywell. The two

	containment high-range radiation monitors listed in Table 3.13.1 shall be Operable when Primary Containment Integrity is required. TS 3.13.G, including Table 3.13.1, is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactor accidents and primary containment integrity are no longer applicable.
3.13.H, High-Range Radioactive Noble Gas Effluent Monitor	This specification requires the capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents and in containment during and following an accident. The high range radioactive noble gas effluent monitors listed in Table 3.13.1 shall be Operable during Power Operation. TS 3.13.H, including Table 3.13.1, is proposed for deletion in the PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactor accidents and primary containment integrity are no longer applicable.

TS SECTION 3.15 – EXPLOSIVE GAS MONITORING INSTRUMENTATION

TS Section 3.15 "Explosive Gas Monitoring Instrumentation," contains LCOs related to the operability of the instrumentation that monitors the hydrogen concentration in the augmented offgas treatment system. Hydrogen is a byproduct of the reactor fission process. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.15 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.15.A, Explosive Gas Instrumentation	This specification requires explosive gas monitoring instrumentation channels shown in Table 3.15.2 to be operable. The instrumentation is provided for monitoring hydrogen below the explosive level in the augmented offgas system downstream from the recombiner. The alarm setpoints are required to be operable when the augmented offgas treatment system is in operation. The alarm/trip setpoints ensure that the hydrogen concentration limits in TS 3.6.F are not exceeded. TS Section 3.15.A, including Table 3.15.2, is proposed for deletion in PDTS. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), hydrogen will no longer be produced in the RCS, a vacuum in the main condenser will not be established, and the condenser offgas will no longer be generated. Therefore, this specification is not required. TS 3.6.F is also being proposed to be eliminated.

TS SECTION 3.17 – CONTROL ROOM HEATING, VENTILATION, AND AIR-CONDITIONING SYSTEM

TS Section 3.17 "Control Room Heating, Ventilation, and Air-Conditioning System," contains the LCOs related to the control room Heating, Ventilation, and Air-Conditioning (HVAC) system. The operability of the control room HVAC system ensures that the control room will remain habitable for operations personnel during a postulated DBA. The operability of the CRE boundary must be maintained to protect the CRE occupants during normal and accident conditions. The CRE and its boundary are defined in the Control Room Envelope Habitability Program. In order for the Control Room HVAC System to be considered operable, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and the CRE occupants are protected from hazardous chemicals and smoke. Since control room HVAC systems A and B do not have high efficiency particulate air (HEPA) filters or charcoal absorbers, the supply fan and dampers for each system minimize the beta and gamma doses to the operators by providing positive pressurization and limiting the makeup and infiltration air into the CRE. For the supply of 100% outside unfiltered air to the CRE under DBA conditions, personnel occupying the control room shall not receive radiation exposure in excess of a 30-day integrated dose of 5 rem total effective dose equivalent (TEDE).

The FHA in the SFP is the only DBA that can occur with the facility in the permanently defueled condition. In Reference 6, Exelon provided an FHA-based analysis using Alternate Source Term methodology. The analysis determined the projected dose due to the drop of a fuel assembly onto other fuel assemblies as a function of time after shutdown. The analysis demonstrates that radiological doses at the exclusion area boundary, low population zone and in the control room from a FHA after 60 days following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system. No equipment is required to mitigate the effects of this event beyond the administrative controls described in Reference 6. The table below describes the specifications in this section.

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once OCNGS dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.17 will not be required and these LCOs (and associated SR in Section 4) will not apply in a permanently defueled condition.

Current OCNGS TS	Basis for Deletion
3.17.A - D	TS 3.17.A requires the control room HVAC system to be operable during all modes of plant operation. TS 3.17.B provides required action when one control room HVAC system is inoperable for reasons other than specification D. TS 3.17.C provides required action when both control room HVAC system is inoperable for reasons other than specification D. TS 3.17.D provides required action when one or both control room HVAC systems are determined inoperable due to an inoperable CRE boundary. TS 3.17 A through D is proposed for deletion in the PDTS since the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted and after 60 days of decay of irradiated fuel following shut down. Additionally, the design basis FHA was assessed for post-cessation of power operations in order to justify the elimination of TS requirements for the operability of control room HVAC systems. The results of the evaluation indicated a 60-day decay time after permanent shutdown to

	meet the regulatory acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183 without credit for HVAC systems. Therefore, based on the permanent shutdown condition of the reactor and the post-shutdown FHA in the SFP analysis, TS 3.17 is proposed for deletion in the PDTS.

TS SECTION 4 – SURVIELLANCE REQUIREMENTS

TS Section 4 describes surveillance requirements (SR) associated with the TS Section 3 LCOs. In accordance with 10 CFR 50.36(c)(3), surveillance requirements are related 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 LCOs will be met.

Since there are no safety limits that apply to OCNGS with the reactor shutdown and defueled, and since there are relatively few remaining LCOs, the number of corresponding surveillance requirements has also been greatly reduced.

Due to the reduced number of LCOs and Surveillance Requirements, OCNGS proposes to combine the LCOs (TS Section 3) with the corresponding Surveillance Requirements (TS Section 4) This format will allow the Surveillance Requirements to be more readily associated with the corresponding LCO. The LCOs and combined Surveillance Requirements (SR) sections will be designated with notation 3/4.#. The proposed format to the LCOs is shown in Attachment 2.

The list below contains a comparison between the provisions of the current OCNGS TS and the proposed PDTS. Each subsection of OCNGS TS Section 4 is discussed in more detail in the tables below.

Current OCNGS TS	Proposed PDTS
4.0 Surveillance Requirement Applicability	Applicable (Proposed New ***3/4.0 Limiting Conditions for Operations and Surveillance Requirement Applicability***)
4.1 Protective Instrumentation	Not Applicable (Proposed New ***3/4.1 Spent Fuel Storage***)
4.2 Reactivity Control	Not Applicable (Proposed ***3/4.2 Radioactive Liquid Storage***)
4.3 Reactor Coolant	Not Applicable
4.4 Emergency Cooling	Not Applicable
4.5 Containment System	Not Applicable
4.6 Radioactive Effluents	Remaining SR 3.6.C Renumbered as 3/4.2
4.7 Auxiliary Electrical Power	Not Applicable
4.8 Isolation Condenser	Not Applicable
4.9 Refueling	Not Applicable
4.10 ECCS Related Core Limits	Not Applicable
4.11 Sealed Source Contamination	Not Applicable
4.12 Alternate Shutdown Monitoring Instrumentation	Not Applicable
4.13 Accident Monitoring Instrumentation	Not Applicable
4.14 Solid Radioactive Waste	Previously Deleted
4.15 Explosive Gas Monitoring Instrumentation	Not Applicable
4.16 Radiological Environmental Surveillance	Previously Relocated to the ODCM
4.17 Control Room Heating, Ventilation and Air Conditioning System	Not Applicable

TS SECTION 4.0, SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

TS Section 4.0 "Surveillance Requirement Applicability," contain the general requirements applicable to all SRs and applies at all times unless otherwise stated in TSs. Surveillance requirements are requirements relating to test or inspection to assure that the necessary quality of systems and components is maintained.

SRs 4.0.1 and 4.0.3 have been revised to reflect the permanently shutdown and defueled condition in the proposed PDTS. SR 4.0.2 has been maintained in its entirety without change. A new SR (4.0.4) is being proposed in the PDTS (see discussion in proposed TS Section 3/4.0). This SR is based on NUREG-1433, *"Standard Technical Specifications General Electric BWR/4 Plants"* (Reference 8) and Draft NUREG-1625, *"Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants"* (Reference 13), which has been modified to reflect the permanently defueled condition.

Current OCNGS TS	Proposed OCNGS TS
TS 4.0.1 - Surveillance requirements shall be met during the modes or other specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in SR 4.0.2. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.	**SR** 4.0.1 – Surveillance requirements shall be met during the ~~modes or other~~ specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in SR 4.0.2. Surveillances do not have to be performed on ~~inoperable equipment or~~ variables outside specified limits.

Basis

TS 4.0.1 establishes the requirement that SRs must be met during the modes or other specified conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits.

TS 4.0.1 is proposed for revision to remove references to operating modes and inoperable equipment. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the reference to operating modes is no longer relevant and is therefore being deleted. Since there are no LCOs for equipment to be operable or in operation in the PDTS, the exception to not perform surveillances on inoperable equipment is no longer needed.

TS 4.0.1 is relocated to proposed PDTS Section 3/4.0 as SR 4.0.1. This revision is editorial in nature.

<table>
<tr><th>Current OCNGS TS</th><th>Proposed OCNGS TS</th></tr>
<tr><td>TS 4.0.2 - If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.</td><td>SR 4.02

If it is discovered that a surveillance <…>

<u>No Change</u>

TS 4.0.2 is relocated to proposed PDTS Section 3/4.0 as SR 4.0.2. This revision is editorial in nature.</td></tr>
<tr><th>Current OCNGS TS</th><th>Proposed OCNGS TS</th></tr>
<tr><td>TS 4.0.3 - Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillance have been met within their specified frequency, except as provided by 4.0.2. When an LCO is not met due to surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.C.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.</td><td>SR 4.0.3 - Entry into a ~~an OPERATIONAL CONDITION or other~~ specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillance has~~have~~ been met within its~~their~~ specified frequency, except as provided by 4.0.2. ~~When an LCO is not met due to surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.C.~~

This provision shall not prevent entry into ~~OPERATIONAL CONDITIONS or~~ other specified conditions in the Applicability that are required to comply with LCO requirements~~ or that are part of a shutdown of the unit~~.</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">TS 4.0.3 establishes the requirements that all applicable SRs must be met before entry into an operational mode or other specified condition in the applicability.

TS 4.0.3 is being modified, such that, the surveillance requirements in proposed PDTS TS 3/4.1 for SFP must be met prior to entry in to the specified condition. The remaining application is not necessary to preclude this and is being eliminated. The revision includes grammatical corrections.

Because 10 CFR 50.82(a)(2) will prohibit operation of the plant or placing fuel in the reactor vessel, the reference to OPERATIONAL CONDITION and shutdown of the unit will no longer be relevant and are being deleted. Additionally, the reference to exceptions and allowances stated in the TS LCO is deleted since these items are not applicable in PDTS.

TS 4.0.3 is relocated to proposed PDTS Section 3/4.0 as SR 4.0.3. This revision is editorial in nature.</td></tr>
</table>

Current OCNGS TS	Basis for Deletion
TS Section 4.1 - Protective Instrumentation	This section specifies the minimum frequency and type of surveillance to be applied to the reactor protection system, reactor isolation, isolation condenser initiation and isolation, primary containment isolation, emergency core cooling actuation, control rod block actuation, offgas system isolation, diesel generator load sequence timers, secondary containment integrity and loss of normal power systems. These surveillance requirements apply to instrumentation and associated devices that initiate reactor scram and other protective functions. Table 4.1.1 and Table 4.1.2 of TS 4.1 specifies testing and frequency requirements for protective instrumentation and trip systems. As discussed in TS Section 3.1 above, the protective functions based on postulated accident scenarios requiring actuation of these systems are no longer necessary at OCNGS and these specifications are proposed for deletion in the PDTS. Additionally, as discussed in the “Basis for Deletion” for Table 3.1.1, Sections J and N, the protective functions associated with the instrumentation identified in Section 4.1, Table 4.1.1, Items 14 and 28, “High Radiation in Reactor Building Operating Floor Ventilation Exhaust instrumentation” and “Loss of Power,” respectively, are no longer required at OCNGS because neither of the two protective functions are required or relied on to prevent the occurrence or mitigate the consequences of a FHA in the SFP. Therefore, based on the above, the surveillance requirements of OCNGS TS 4.1 and Table 4.1.1 are also proposed for deletion.
TS Section 4.2 - Reactivity Control	This section specifies the minimum frequency and type of surveillance to be applied to reactivity limits and control systems to ensure operability. These surveillance requirements apply to reactivity limitations and systems required during power operation of the reactor or core alterations. As discussed in TS Sections 3.2.A through 3.2.D, since OCNGS has permanently ceased power operation and is no longer authorized to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), these systems will no longer be required at OCNGS and these specifications are proposed for deletion in the PDTS. Therefore, the surveillance requirements of TS 4.2 are also proposed for deletion.
TS Section 4.3 - Reactor Coolant	This section specifies the minimum frequency and type of surveillance to be applied to the primary system boundary. These specifications were designed to ensure the reactor coolant system is maintained within design limits to prevent a breach of the system that could result in a release of fission products to the environment. These surveillance requirements include reactor coolant system sampling and leakage monitoring, reactor vessel material surveillance, inservice testing and inspection and safety valve setpoints. As discussed in TS Sections 3.3.A through 3.3.H, since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor, pursuant to 10 CFR 50.82(a)(2), the protection of the reactor coolant pressure boundary is no longer required and these specifications are proposed for deletion in the PDTS. Therefore, the surveillance requirements are also proposed for deletion.

TS Section 4.4 - Emergency Cooling	This section specifies the minimum frequency and type of surveillance applied to emergency core cooling and containment cooling systems. These systems are designed to ensure the reactor core is cooled to minimize fuel clad failure and primary containment is cooled to prevent a breach of primary containment which could result in a release of fission products to the environment. As discussed in TS Sections 3.4.A through 3.4.F, since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor, pursuant to 10 CFR 50.82(a)(2), core and containment cooling systems are no longer required and these specifications are proposed for deletion in the PDTS. Therefore, surveillance requirements are also proposed for deletion.
TS Section 4.5 - Containment System	Primary containment systems and primary containment isolation valves (addressed in LCO 3.5.A) are designed to minimize the release of fission products to the environment after a design basis accident. These specifications ensure the integrity of the primary containment. The surveillance requirements in these TS sections (4.5.A through 4.5.F and 4.5.I through 4.5.N) are required to meet primary containment integrity. As discussed in TS Section 3.5.A, OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2): therefore, primary containment integrity and primary containment isolation valve operability are no longer required. Secondary containment and the SGTS (addressed in LCO 3.5.B) mitigates the release of fission products to the environment after a reactor-related accident and also function to mitigate the release of fission products to the environment in the event of a fuel handling accident. The surveillance requirements in these sections (4.5.G and 4.5.H) are required to meet secondary containment integrity. As discussed in the “Fuel Handling Accident Analysis for the Permanently Defueled Condition” in Section 2.0, “Detailed Description and Basis for the Changes,” and TS Section 3.5.B, Exelon recently reanalyzed the FHA in the SFP, assuming a ground level release without credit for secondary containment integrity and operation of the SGTS. Radiological consequences at the exclusion area boundary and low population zone are well within 10 CFR Part 50.67 limits 60 days after final shutdown. Since OCNGS will no longer rely on secondary containment integrity to mitigate the radiological consequences of a fuel handling accident, these systems are no longer required to be operable nor are they required to perform any safety function. Therefore, LCO 3.5.B is proposed for deletion in the PDTS. Since the LCOs in TS Section 3.5 are proposed for deletion in the PDTS, the surveillance requirements are also proposed for deletion.
TS Section 4.6 - Radioactive Effluent	This section specifies the minimum frequency and type of surveillance applied to the release of radioactive liquids and gases. These systems and specifications are designed to monitor radioactive releases to ensure all releases are monitored and do not exceed maximum release requirements.

	As discussion above, TS 3.6.C is being maintained and renumbered in the PDTS as TS 3/4.2. The corresponding SR is also being maintained and included in TS 3/4.2. The proposed TS 3/4. 2 LCO and SR is shown in the discussion above and in Attachment 2. Also, as discussed above, TS Sections 3.6.A, 3.6.D, 3.6.E, and 3.6.F are not applicable in the permanently defueled condition and are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.7 - Auxiliary Electrical Power	This section specifies the minimum frequency and type of surveillance applied to the AC and DC electrical power systems during operation. As discussed in TS Sections 3.7.A, 3.7.B and 3.7.C, the design basis accidents that require power for the emergency safeguards equipment systems supplied by the auxiliary electrical system will no longer be applicable; therefore, the electrical system specifications are no longer required. These specifications are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.8 - Isolation Condenser	This section specifies the frequency and type of surveillance to ensure operability of the isolation condenser system. As discussed in TS Sections 3.8.A through 3.8.F, the isolation condensers will not be required in the permanently defueled condition. These specifications are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.9 - Refueling	This section specifies frequency and type of surveillance applied to refueling systems. These specifications are designed to ensure core reactivity is within the capability of the control rods to prevent criticality during refueling. As discussed in the TS Section 3.9.A through 3.9.G, OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor. Therefore, refueling interlocks and core monitoring provisions are no longer required These specifications are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.10 - ECCS Related Core Limits	This section specifies frequency and type of surveillance applied to power distribution limits. As discussed in TS Sections 3.10.A through 3.10.C, since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor, pursuant to 10 CFR 50.82(a)(2), power distribution limits are no longer applicable. These specifications are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.11 - Sealed Source Contamination	TS 4.11 applies to each licensed sealed source containing radioactive material either in excess of 100 microcurics of beta and/or gamma emitting materials or 5 microcuries of alpha emitting material. This requirement ensures that the total body or individual organ irradiation does not exceed

	allowable limits in the event of ingestion or inhalation of the probable leakage from a source material. These specifications are proposed from deletion in the PDTS. This requirement is not credited in any safety analysis and does not meet any of the criteria in 10 CFR 50.36(c)(2)(ii). Therefore, this specification can be deleted from the PDTS. Licensed sealed sources are controlled under a licensee controlled program.
TS Section 4.12 - Alternate Shutdown Monitoring Instrumentation	The surveillance requirements in this section ensure that alternate shutdown monitoring instrumentation is operable in order to support reactor power operation and shutdown in the event the control room is uninhabitable. As discussed in TS Sections 3.12.A and 3.12.B, the specifications apply only when the plant is at power operation and when reactor coolant temperature is above 212 degrees F. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the need to achieve and maintain hot shutdown are no longer applicable. These specifications are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements contained in this section are also proposed for deletion.
TS Section 4.13 - Accident Monitoring Instrumentation	This section specifies frequency and type of surveillance applied to accident monitoring instrumentation. As discussed in the TS Sections 3.13.A through 3.13.H, the instrumentation contained in these specifications is required to be operable during power operation or when primary containment integrity is required to monitor the course of reactor accidents. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactor accidents and primary containment integrity are no longer applicable. These specifications are proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.14 - Solid Radioactive Waste	TS Section 4.14 was previously deleted by License Amendment No. 166. and serves as a placeholder. Placeholder specification titles are being proposed for deletion. This change is editorial in nature.
TS Section 4.15 - Explosive Gas Monitoring Instrumentation	This section contains surveillance requirements related to explosive gas monitoring instrumentation. As discussed in OCNGS TS Section 3.15.A, offgas containing hydrogen will no longer be generated. This specification is proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.
TS Section 4.16 - Radiological Environmental Surveillance	TS Section 4.16 was previously relocated to the ODCM by License Amendment No. 166 and serves as a placeholder. Placeholder specification are being proposed for deletion. This change is editorial in nature.

TS Section 4.17 - Control Room Heating, Ventilation, and Air-Conditioning System	This section contains surveillance requirements related to the control room HVAC system. The specifications contained in this section ensure the operability of the control room HVAC system. As discussed in OCNGS TS Section 3.17.A through 3.17.D, this system assures that the control room will remain habitable during design basis reactor accidents. Since OCNGS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), reactor accidents are no longer possible. Additionally, the design basis FHA was assessed for post-cessation of power operations in order to justify the elimination of TS requirements for the operability of control room HVAC systems. The results of the evaluation indicated a 60-day decay time after permanent shutdown to meet the regulatory acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183 without credit for HVAC systems. Therefore, based on the permanent shutdown condition of the reactor and the post-shutdown FHA in the SFP analysis, TS 3.17 is proposed for deletion in the PDTS. Therefore, the corresponding surveillance requirements are also proposed for deletion.

TS SECTION 5 – DESIGN FEATURES

The existing TS Section 5 "Design Features," contains descriptions and requirements for those features of the facility such as materials of construction and geometric arrangements which, if altered or modified, could have a significant effect on safety and are not covered in the previous sections of the TS.

Because the OCNGS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the design features that do not apply in a defueled condition are being proposed for deletion. TS 5.1.A and TS 5.3 (renumbered to TS 5.2) will remain applicable with the reactor permanently defueled. As such, these TS sections are being retained to reflect a permanently defueled condition.

Current OCNGS TS	Proposed OCNGS TS
TS 5.1.B The reactor building, standby gas treatment system and stack shall comprise a secondary containment in such fashion to enclose the primary containment in order to provide for controlled elevated release of the reactor building atmosphere under accident conditions.	(Deleted with TS identifier editorially removed)
TS 5.2 Containment A. The primary containment <…>	Delete (TS 5.3 renumbered as TS 5.2)

Basis

TS Section 5.1.B is proposed for deletion. As discussed in TS Section 3.5, the SBGT system is proposed for deletion. Once OCNGS has permanently defueled, the potential for a release of fission products to the environment in the event of a LOCA no longer exist. Therefore, maintaining primary and secondary

containment integrity is no longer applicable. The FHA in the SFP accident analysis shows that secondary containment integrity and the SGTS for mitigation of radiological releases is no longer necessary.

TS Section 5.2 Containment, provides references to principal design parameters and applicable design codes for the primary containment, the secondary containment and applicable codes, and design standards for penetrations to the primary containment and piping passing through such penetrations.

Because the OCNGS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), this TS section will not apply in a defueled condition and is being proposed for deletion.

Current OCNGS TS	Proposed OCNGS TS
TS 5.3 AUXILIARY EQUIPMENT 5.3.1 Fuel Storage A. The fuel storage facilities are designed and shall be maintained with a K-effective equivalent to less than or equal to 0.95 including all calculational uncertainties. B. Deleted C. Deleted D. The temperature of the water in the spent fuel storage pool, measured at or near the surface, shall not exceed 125°F. E. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 3035.	TS 5.~~3~~**2** ***SPENT FUEL STORAGE***~~AUXILIARY EQUIPMENT~~ 5.~~3~~**2**.1 ***Spent*** Fuel Storage A. The ***spent*** fuel storage facilities are designed and shall be maintained with a K-effective equivalent to less than or equal to 0.95 including all calculational uncertainties. ~~B. Deleted~~ ~~C. Deleted~~ ~~D~~***B***. The temperature of the water in the spent fuel storage pool, measured at or near the surface, shall not exceed 125°F. ~~E~~***C***. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 3035.

Basis

TS Sections 5.3.A, D, and E describe and provide the requirements regarding prevention of criticality of spent fuel, temperature limitation of SFP water, and SFP capacity limitations. TS Sections 5.3.1 and 5.3.1.A are proposed to be modified to clarify that the requirements are applicable to the spent fuel storage since there will be no new fuel storage maintained after the permanent shutdown and defueling and therefore the requirements apply only to the spent fuel storage design. TSs 5.3.1 D and E are being retained as-is in the proposed PDTS. These specifications are being renumbered to TS 5.2.A-C.

TS SECTION 6 – ADMINISTRATIVE CONTROLS

The existing TS Section 6 Administrative Controls, contains provisions relating to organization and management, procedures, recordkeeping, review and audit, programs, and reporting necessary to assure operation of the facility in a safe manner. The NRC approved TS License Amendment No. 290 (Reference 3) that revised and removed certain requirements from Section 6.0 that are not applicable to the facility in a permanently defueled condition. Specifically, the amendment will revise TS Section 6.1, "Responsibility"; TS Section 6.2, "Organization"; TS Section 6.3, "Facility Staff Qualifications"; TS Section 6.6, "Reportable Event Action"; TS Section 6.7, "Safety Limit Violation"; TS Section 6.8, "Procedures and Programs"; and TS Section 6.9, "Reporting Requirements" to reflect the staffing and training requirements for operating staff when the facility is permanently defueled and other minor administrative changes.

License Amendment No. 290 will be effective upon the submittal of the certifications required by 10 CFR 50.82(a)(1)(i) and (ii), and will be implemented after OCNGS is permanently defueled. The additional

changes requested in this proposed LAR will become effective 60 days following permanent shutdown of OCNGS to allow for the 60-day decay period established in the FHA in the SFP analysis.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, several of the TS Section 6.0 Specifications are no longer applicable. Therefore, the administrative controls that do not apply in a defueled condition are being proposed for deletion.

Current OCNGS TS	Basis for Change/Deletion
TS 6.8.1.a. The procedures applicable<…> as referenced in the QATR.	It is proposed to revise this specification to remove "QATR" and replace it with "Decommissioning Quality Assurance Program." The QATR is the quality assurance topical report which is the Exelon QA program for its operating reactors. The QATR is being replaced by a QA program for decommissioning plants.
TS 6.8.4.a, Radioactive Effluents Controls Program and TS 6.8.4.b, Radiological Environment Monitoring Program	The proposed changes to these specifications reformats previously defined terms from upper case to lowercase letters. As discussed in TS Sections 1.35 and 1.38, the terms "MEMBERS OF THE PUBLIC," "UNRESTRICTED AREA," and "SITE BOUNDARY" are proposed for deletion as defined terms in this license amendment request. The standard convention of indicating defined terms in all capital letters has been adopted in the PDTS. Therefore, since these terms are no longer defined within the context of the PDTS, they are being reformatted to lowercase. This change is editorial in nature. There are no other proposed changes to these TSs.
TS 6.8.4.b, Radiological Environment Monitoring Program 2. Land Use Census	The requirement for a Land Use Census and participation in an Interlaboratory Comparison Program are proposed for deletion from PDTS. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, OCNGS will no longer be authorized to operate or retain fuel in the reactor vessel. An analysis of the FHA in the SFP indicated that radiological doses at the EAB and LPZ are within allowable limits of 10 CFR 50.67 after a 60-day fuel decay period following permanent reactor shutdown. There will no longer be a need to modify the radiological monitoring program for use of areas at or beyond the site boundary. The land use census is proposed for deletion from the PDTS. The land use census is currently controlled by the ODCM with report made to the NRC in the Annual Radiological Environmental Operating Report (AREOR). The land use census satisfies the requirements of 10CFR50 Appendix I Section IV.B.3. The land use census will be retained in the ODCM.
3. Interlaboratory Comparison Program	The Interlaboratory Comparison Program is proposed for deletion from PDTS. This program is currently controlled in the ODCM which directs reports be made to the NRC in the AREOR. The Interlaboratory Comparison Program satisfies the requirements of 10CFR50 Appendix I Section IV.B.2. This program will be retained in the ODCM.
TS 6.8.5, Station Battery Monitoring and Maintenance Program	This program was established to provide for Station battery restoration and maintenance. As discussed in the bases for the deletion of TS 3.7.D, the revised analysis for the FHA in the SFP, which is the only DBA applicable to the permanently defueled condition does not rely on batteries for accident mitigation and TS 3.7.D is proposed for deletion.

	Therefore, the "Station Battery Monitoring and Maintenance Program," is correspondingly being proposed for deletion in the PDTS.
TS 6.9.1, Routine Reports a. DELETED b. DELETED c. DELETED d. Radioactive Effluent Release Report e. Annual Radiological Environmental Operating Report f. DELETED Basis: 6.9.1.e - RELOCATED TO THE ODCM.	The proposed change to TS 6.9.1.a-f is to eliminate term "DELETED" and renumber item "d" as "a" and item "e" as "b". This change is editorial in nature. a. Radioactive Effluent Release Report b. Annual Radiological Environmental Operating Report It is also proposed to eliminate "Basis: 6.9.1.e – Relocated to the ODCM" since it does not provide any information that contributes to the safe storage and management of the spent fuel in the SFP. This change is editorial in nature.
TS 6.9.2 DELETED	TS Section 6.9.2 was deleted in Amendment No. 290 and serves as a placeholder. Placeholder specification are being proposed for deletion. This change is editorial in nature.
TS 6.9.3, Unique Reporting Requirements a. Material Radiation Surveillance Specimen Report	TS 6.9.3 is proposed for deletion in its entirety. As discussed below, all specifications requiring Special Reports to the NRC are being proposed for deletion in this LAR or have been relocated to the ODCM. The following specifications are being proposed for deletion from the PDTS. a. This specification addresses the reporting of the test results of material surveillance specimens and neutron flux monitors that were installed in the reactor vessel adjacent to the wall at the midplane of the active core. This report is associated with SR 4.3.A, which as discussed in TS Section 4.3, and is proposed to be deleted. Therefore, this report is no longer relevant and is proposed to be deleted.
c. Results of required leak tests performed on sealed sources <…>	c. This specification addresses the reporting of the presence of removable contamination on sealed sources if greater than 0.005 microcuries. This report is associated with SR 4.11, which as discussed in TS Section 4.11, and is proposed to be deleted. Therefore, this report is no longer relevant and is proposed to be deleted.
e-j. Pursuant to the ODCM.	e-j. These specifications were relocated to the ODCM in Amendment No. 166 and this notation serves as a placeholder which is proposed for deletion.
k. Records of results of analyses required by the Radiological Environmental Monitoring Program.	k. This specification is proposed for deletion as the records are reported in the Annual Radiological Environmental Operating Report required in TS 6.9.1.e (proposed TS 6.9.1.b).
l. Failures and challenges to Relief and Safety Valves <…>	l. This specification addresses the reporting of failures and challenges to relief and safety valves which do not constitute a Licensee Event Report (LER). Since in the permanently defueled condition, there are no safety or relief valves that are required to operate to mitigate the

m Plans for compliance with standby liquid control <…> n. Inoperable high range radioactive noble gas effluent monitor	consequences of the FHA in the SFP, this reporting requirement is no longer relevant and is proposed to be deleted. m. This specification addresses the reporting non-compliance with the boron requirements with the standby liquid control system. This report is associated with the actions required in TS 3.2.C.3(b), TS 3.2.C.3.(e)(1), and SR 4.2.E.5which as discussed in the TS Sections 3.2.C and 4.2 are proposed to be deleted. Therefore, this report is no longer relevant and is proposed to be deleted. n. This specification addresses the reporting if the high range radioactive noble gas effluent monitor is inoperable for greater than 7-days. This report is associated with TS 3.13.H which as discussed in the TS Section 3.13.H is proposed to be deleted. Therefore, this report is no longer relevant and is proposed to be deleted.
TS 6.10, Record Retention	The requirements for Record Retention are proposed to be deleted from PDTS on the basis that they can be adequately address by the Quality Assurance Program (10 CFR 50, Appendix B, Criterion XVII) and because provisions relating to record keeping do not assure safe operation of a facility in a permanently defueled condition. Facility operations in a defueled facility are performed in accordance with approved written procedures. Facility records document appropriate station activities. Retention of the records provides document retrievability for review of compliance with requirements and regulations. Post-compliance review of records does not assure operation of the facility in a safe manner as activities described in these documents have already been performed. Numerous other regulations such as 10 CFR 20, Subpart L, and 10 CFR 50.71 also require retention of certain records related to operation of the facility. Thus, Record Retention will be maintained by the Decommissioning Quality Assurance Program.
TS 6.11, Radiation Protection Program	This program requires procedures to be prepared for personnel radiation protection consistent with the requirements of 10 CFR 20. These procedures are developed to ensure nuclear plant personnel safety and have no impact on nuclear safety. Additionally, nuclear plant personnel are not 'members of the public.' Thus, the principal operative standard in Section 182a of the Atomic Energy Act: 'health and safety of the public' does not apply. The Radiation Protection Program administrative control is proposed to be deleted from the PDTS. The program is not necessary to assure operation of the facility in a safe manner and can be relocated from the TS to the UFSAR. The requirement to have procedures to implement Part 20 and the requirement for periodic review of these procedures is addressed under 10 CFR 20 Subpart B – Radiation Protection Programs.
TS 6.14, Environmental Qualification	This specification addresses the NRC Order for codifying the documentation requirements for the environmental qualification of safety-related electrical equipment for OCNGS (Reference **Error! Reference source not found.**). Requirements for environmental qualification of safety-related electrical equipment, including documentation requirements were later codified in 10 CFR 50.49. As stipulated in 10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants (NPP) is applicable for all NPPs except

	plants which have submitted the certifications required under § 50.82(a)(1). Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, as stipulated in 50.49, this program is no longer required. Therefore, TS 6.14 Environmental Qualifications (EQ) is proposed for deletion from the PDTS.
TS 6.15, Integrity of Systems Outside Containment	This program was established to provide controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, specifically Core Spray, Containment Spray, Reactor Water Cleanup, Isolation Condenser, and Shutdown Cooling. Once the plant is permanently shut down and defueled, there will no longer be any transient or accident conditions associated with primary coolant sources that would require these systems nor would there be any potential for leakage of highly radioactive fluids from them. Therefore, TS 6.15is proposed for deletion in the PDTS.
TS 6.16, Iodine Monitoring	The Iodine Monitoring Program provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program was developed to minimize radiation exposure to plant personnel post-accident. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted and after 60 days of decay following shut down, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel. Therefore, accident conditions associated with the reactor coolant system will no longer apply to the permanently shut down and defueled facility. The design basis FHA in the SFP was assessed for post-cessation of power operations. Airborne iodine concentration will no longer have an impact on facility operation. Thus, TS 6.16 is proposed for deletion in the PDTS.
TS 6.18, Process Control Plan	The specification for the Process Control Plan (PCP) is proposed for deletion. The PCP implements the requirements of 10 CFR Parts 20, 61 and 71, and 49 CFR 171-172. The actions of TS 6.18 are required by regulation and it is not necessary to restate the requirements in the PDTS. The PCP is conducted under standard procedures with revisions approved by facility processes and program changes are reported to NRC by the Annual Radioactive Effluent Report. Thus, TS 6.18 is proposed for deletion in the PDTS.
TS 6.19, Offsite Dose Calculation Manual	TS 6.19 is proposed to be retained with the following revision. Sub-paragraph "a", states, "The ODCM shall be approved by the Commission prior to implementation." This sub-paragraph is proposed for deletion since this action has been completed. Sub-paragraphs "b" and "c" are proposed to be renumbered as "a" and "b". This action is administrative in nature.
TS 6.20, Major Changes to Radioactive Waste Treatment Systems	TS Section 6.20 was previously deleted; however, the title was not deleted. This proposed change deletes the title and replaces it with DELETED. This change is editorial in nature.

TS 6.22, Control Room Envelope Habitability Program	This program ensures CRE habitability is maintained such that, with an operable HVAC system, the CRE occupants can control the reactor safely under normal and maintain it in a safe condition following a postulated accident or event. Equipment control from the control room is no longer required to mitigate any of the remaining postulated accidents or events. With the plant in a permanently defueled state, the postulated accidents analyzed in UFSAR Chapter 15 are no longer credible, with the exception of the FHA in the SFP, as discussed in the Fuel Handling Accident Analysis for the Permanently Defueled Condition section of this enclosure which does not credit or require the use of the control room for mitigation. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted and after 60 days of decay following shut down, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel. The design basis FHA in the SFP was assessed for post-cessation of power operations in order to justify the elimination of TS requirements for the operability of control room ventilation systems. As previously discussed, TSs 3/4.17 is proposed for deletion in the PDTS.
TS 6.23, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)	This program contains RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates utilized in TS Section 3.3. These limits were established to address reactor vessel brittle fracture and neutron embrittlement of the reactor vessel. Since the OCNGS license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the RCS will remain depressurized. TS Sections 3.3 is being proposed for deletion in this license amendment request and the information associated with the PTLR will no longer be applicable. Thus, TS 6.23 is proposed for deletion in the PDTS.
TS 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. The requirements regarding the Surveillance Frequency Control Program (SFCP) are proposed for deletion in the PDTS. The remaining TS LCOs proposed for this PDTS contains two surveillance requirements (SR). Therefore, no further need to maintain this program exists and it can be eliminated. Thus, TS 6.24 is proposed for deletion in the PDTS.
TS 6.25, Snubber Inspection Program	This program established the examination, testing and service life conditions for dynamic restraints (Snubbers) in accordance with 10 CFR 50.55a inservice inspection requirements for supports. TS 6.25 is being proposed for deletion in its entirety. Since 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, all systems associated with snubbers are no longer required to be operable. As such, the requirements provided by TS 6.25 are no longer needed.

The proposed changes are shown on the marked-up OCNGS TS pages included as Attachment 2.

3.0 REGULATORY EVALUATION

3.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Exelon has determined that the proposed changes do not require any exemptions or relief from regulatory requirements.

10 CFR 50.82 *"Termination of license."*

"(a) For power reactor licensees —

(1) (i) When a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of § 50.4(b)(8);

(ii) Once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of § 50.4(b)(9) and; <…>

(2) Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

By letter dated January 7, 2011 (Reference 1), Exelon provided formal notification to the NRC pursuant to 10 CFR 50.82(a)(1)(i) of Exelon's contingent determination to permanently cease operations at OCNGS by December 31, 2019.

10 CFR 50.36 *"Technical specifications."*

In 10 CFR 50.36, *"Technical specifications,"* the NRC established its regulatory requirements related to the content of TSs. In doing so, the NRC placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the NRC noted that applicants were expected to incorporate into their TSs "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." (Statement of Consideration, "Technical Specification for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968).

Pursuant to 10 CFR 50.36, TS are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports. However, the rule does not specify the particular requirements to be included in a plant's TS.

Section 50.36 of 10 CFR provides four criteria to define the scope of equipment and parameters to be included in the TS LCOs. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the reactor coolant system pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during design bases accidents or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those structures, systems, and components (SSCs) shown to be significant to

public health and safety. These criteria, which were subsequently codified in changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953), also pertain to the TS requirements for safe storage of spent fuel. A general discussion of these considerations is provided below.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for *"installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary."* Since no fuel will be present in the reactor or the reactor coolant system pressurized at the OCNGS facility following permanent defueling, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for *"a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."* The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some DBAs which continue to apply to a facility authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a facility with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The applicable DBA for OCNGS in the permanently defueled condition, the FHA in the SFP, is discussed within this proposed amendment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for an *"SSC that is part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."* The intent of this criterion is to capture into the TSs only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that will continue to apply to OCNGS, there is still a single credible DBA (FHA in the SFP) that will continue to apply. The scope of the FHA in the SFP is discussed elsewhere in more detail within this proposed amendment.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for an *"SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."* The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. All of the accident sequences that previously dominated risk at OCNGS will no longer be applicable once the reactor is in the permanently shut down and defueled condition.

10 CFR 50.36(c)(5) *Administrative Controls*. *"Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."*

The particular administrative controls to be included in the TS generally are requirements the NRC deems necessary to support the safe operation of a facility that are not already covered by other regulations. Although 10 CFR 50.36 includes these requirements, they are predominately specified in support of an operating plant. Once OCNGS is in a

permanently shutdown and defueled condition, certain administrative controls described in the TS will no longer be required and can be deleted or modified as reflected in this LAR.

10 CFR 50.36(c)(6) *Decommissioning. "This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis."*

As noted above, by letter dated January 7, 2011 (Reference 1), Exelon provided formal notification to the NRC pursuant to 10 CFR 50.82(a)(1)(i) of Exelon's determination to permanently cease operations at OCNGS by December 31, 2019. Upon submittal of the final certification that fuel has been permanently removed from the OCNGS reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii), OCNGS will no longer be licensed to operate. The proposed amendment deletes the portions of the previous OCNGS TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shutdown condition.

10 CFR 50.46, *"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."*

"(a)(1)(i) <…> *This section does not apply to a nuclear power reactor facility for which the certifications required under 10 CFR 50.82(a)(1) have been submitted."*

10 CFR 50.48 *"Fire Protection."*

"(f) Licensees that have submitted the certifications required under § 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard). A fire protection program that complies with NFPA 805 shall be deemed to be acceptable for complying with the requirements of this paragraph.

> *(1) The objectives of the fire protection program are to--*
>
> *(i) Reasonably prevent these fires from occurring;*
>
> *(ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and*
>
> *(iii) Ensure that the risk of fire-induced radiological hazards to the public, environment and plant personnel is minimized.*
>
> *(2) The licensee shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning.*
>
> *(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities."*

In 10 CFR 50.48(f), the NRC established the requirement for maintaining a fire protection program once a licensee has submitted the certifications required under 10 CFR 50.82(a)(1). Since the initial certification has been submitted pursuant to 10 CFR 50.82(a)(1)(i) (Reference 1) and once the final certification required by 10 CFR

50.82(a)(1)(ii) has been submitted, the requirements of 10 CFR 50.48(f) will be in full effect.

10 CFR 50.51, *"Continuation of license."*

"(b) Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the NRC notifies the licensee in writing that the license is terminated. During such period of continued effectiveness the licensee shall--

> *(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and*
>
> *(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."*

Exelon will continue to conduct activities in accordance with the license until the NRC notifies Exelon in writing that the license is terminated.

10 CFR 50.62, *"Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."*

"(a) *Applicability. The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted."*

10 CFR 50.67, *"Accident source term."*

"(a) Applicability. The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, and holders of renewed licenses under part 54 of this chapter whose initial operating license was issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) Requirements. (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents1 previously analyzed in the safety analysis report.

> *(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:*
>
> *(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)2 total effective dose equivalent (TEDE).*
>
> *(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).*
>
> *(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."*

10 CFR 50.2 *"Definitions."*

"Certified fuel handler means, for a nuclear power reactor facility, a non-licensed operator who has qualified in accordance with a fuel handler training program approved by the Commission."

"Responsible officer means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or other individual in the organization of a corporation, partnership, or other entity who is vested with executive authority over activities subject to this part."

"Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable."

By letter dated September 6, 2016 (Reference 2), the NRC approval of a Certified Fuel Handler training program for OCNGS.

3.2 Precedent

The proposed changes are consistent with the intent of the license and accompanying PDTS issued to the following facilities that have been permanently shutdown and defueled: (1) Vermont Yankee Nuclear Power Station, for which an amendment was issued on October 7, 2015 (Reference 9); (2) Kewaunee Power Station, for which an amendment was issued on February 13, 2015 (Reference 10); (3) San Onofre Nuclear Generating Station, Units 2 and 3, for which an amendment was issued on July 17, 2015 (Reference 11); and (4) Crystal River Nuclear Plant, Unit 3, for which an amendment was issued on September 4, 2015 (Reference 12).

3.3 No Significant Hazards Consideration (NSHC)

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon), proposes an amendment to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. DPR-16 for Oyster Creek Nuclear Generation Station (OCNGS).

The proposed amendment would revise the RFOL and the associated TS to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

The proposed changes would revise and remove certain requirements contained within the RFOL and TS, and remove the requirements that would no longer be applicable once it has been certified that all fuel has permanently been removed from the OCNGS reactor pursuant to 10 CFR 50.82(a)(1)(ii). Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for OCNGS no longer will authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The proposed changes to the RFOL and TS being proposed for deletion or revision are in

accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages and sections, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS table of contents is also accordingly revised.

The existing OCNGS TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled plant.

Exelon has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed changes would not take effect until OCNGS has permanently ceased operation, entered a permanently defueled condition, and at least 60 days of irradiated fuel decay time after reactor shutdown. The proposed changes would revise the OCNGS RFOL and TS by deleting or modifying certain portions of the TS that are no longer applicable to a permanently shutdown and defueled facility. This change is consistent with the criteria set forth in 10 CFR 50.36 for the contents of TS.

   Chapter 15 of the OCNGS Updated Final Safety Analysis Report (UFSAR) described the design basis accident (DBA) and transient scenarios applicable to OCNGS during power operations. The analyzed accidents that remains applicable to OCNGS in the permanently shut down and defueled condition is a Fuel Handling Accident (FHA) in the SFP (a dropped fuel assembly onto the top of the core will no longer be applicable) and the Postulated Radioactive Tank Failure and Release of Radioactive Liquid Waste while radioactive liquids are still present. The FHA is the remaining accident with radiological consequences and has been revised for the permanently shutdown and defueled condition. The liquid tank accidents analysis remains bounding and unchanged; therefore, is not discussed further in this NSHC evaluation.

   Once the reactor is in a permanently defueled condition, the spent fuel pool (SFP) and its cooling systems will be dedicated only to spent fuel storage. In this condition, the spectrum of credible accidents will be much smaller than for an operational plant. Once the certifications are docketed by OCNGS pursuant to 10 CFR 50.82(a)(1), and the consequent removal of authorization to operate the reactor or to place or retain fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2), the majority of the accident scenarios previously postulated in the UFSAR will no longer be possible and will be removed from the UFSAR under the provisions of 10 CFR 50.59.

The deletion of TS definitions and rules of usage and application, that will not be applicable in a defueled condition, has no impact on facility structures, systems, and components (SSCs) or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shutdown and defueled status of OCNGS has no impact on the remaining applicable DBA. The removal of LCOs or SRs that are related to only the operation of the nuclear reactor or to only the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the defueled mode. The safety functions involving core reactivity control, reactor heat removal, reactor coolant system inventory control, and containment integrity are no longer applicable at OCNGS as a permanently defueled plant. The analyzed accidents involving damage to the reactor coolant system, main steam lines, reactor core, and the subsequent release of radioactive material will no longer be possible at OCNGS.

After OCNGS permanently ceases operation, the future generation of fission products will cease and the remaining source term will decay. The radioactive decay of the irradiated fuel following shutdown of the reactor will have reduced the consequences of the FHA in the SFP below those previously analyzed. The relevant parameter (water level) associated with the fuel pool provides an initial condition for the FHA analysis and is included in the PDTS.

The SFP water level and spent fuel storage TSs are retained to preserve the current requirements for safe storage of irradiated fuel. SFP cooling and makeup related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since there will be sufficient time to effect repairs, establish alternate sources of makeup flow, or establish alternate sources of cooling in the event of a loss of cooling and makeup flow to the SFP.

The deletion and modification of provisions of the administrative controls do not directly affect the design of SSCs necessary for safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the fuel pool. The changes to the administrative controls are administrative in nature and do not affect any accidents applicable to the safe management of irradiated fuel or the permanently shutdown and defueled condition of the reactor.

The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition will be the only operation allowed, and therefore bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed changes to delete and/or modify certain TS have no impact on facility SSCs affecting the safe storage of spent irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of spent irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor or only to

the prevention, diagnosis, or mitigation of reactor related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shutdown and defueled and OCNGS will no longer be authorized to operate the reactor.

The proposed deletion of requirements of the OCNGS RFOL and TS do not affect systems credited in the accident analysis for the FHA in the SFP at OCNGS. The proposed RFOL and PDTS will continue to require proper control and monitoring of safety significant parameters and activities.

The TS regarding SFP water level and spent fuel storage is retained to preserve the current requirements for safe storage of irradiated fuel. The restriction on the SFP water level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA.

The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding and spent fuel cooling). Since extended operation in a defueled condition will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed changes involve deleting and/or modifying certain TS once the OCNGS facility has been permanently shutdown, defueled, and at least 60 days of irradiated fuel decay time after reactor shutdown. As specified in 10 CFR 50.82(a)(2), the 10 CFR 50 license for OCNGS will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel following submittal of the certifications required by 10 CFR 50.82(a)(1). As a result, the occurrence of certain design basis postulated accidents associated with reactor operation is no longer considered credible. The only remaining credible accidents are a FHA and the Postulated Radioactive Releases Due to Liquid Radwaste Tank Failures. The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses that impact either accident.

   The proposed changes are limited to those portions of the RFOL and TS that are not related to the safe storage of irradiated fuel. The requirements that are proposed to be revised or deleted from the OCNGS RFOL and TS are not credited in the existing accident analysis for the remaining applicable postulated accidents; and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated design basis accidents involving the reactor will no longer be possible because the reactor will be permanently shutdown and defueled and OCNGS will no longer be authorized to operate the reactor.

   Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above, Exelon concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and,

accordingly, a finding of no significant hazards consideration is justified.

3.4 <u>Conclusion</u>

In conclusion, based on the considerations discussed above: 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment involves deleting or modifying certain TS Limiting Conditions of Operation and administrative controls in support of proposed decommissioning efforts to reflect the permanently shutdown and defueled condition at OCNGS. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

In addition, the proposed changes involve changes to recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

5.0 REFERENCES

1. Letter from Keith R. Jury, Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission - "Permanent Cessation of Operations at Oyster Creek Nuclear Generating Station," dated January 7, 2011 (ML110070507)
2. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station; Clinton Power Station, Unit No. 1; and Quad Cities Nuclear Power Station, Units 1 And 2 – Approval of Certified Fuel Handler Training and Retraining Program (CAC NOS. MF8109, MF8138, MF8139, AND MF8140)," dated September 6, 2016 (ML16222A787)
3. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC) – "Oyster Creek Nuclear Generating Station - Issuance of Amendment Regarding Changes to the Administrative Controls Section of the Technical Specifications (CAC NO. MF8108)," dated March 7, 2017 (ML16235A413)
4. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station – Issuance of Amendment RE: Deletion of Facility Operating License Conditions Related to Decommissioning Trust Provisions (CAC No. MF9293)," dated June 23, 2017 (ML17067A042)
5. Letter from David P. Helker (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "License Amendment Request Regarding Revision to Cyber Security Plan Milestone 8 Completion Date," dated April 10, 2017 (ML17100A844).

6. C-1302-226-E310-460, EAB, LPZ, and CR Dose Due to Fuel Handling Accident (FHA) -Post Cessation of Power Operations, dated August 9, 2017
7. NUREG-1875, "Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station," issued April 2007 (ML071290023 & ML071310246)
8. NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," Revision 4 (ML12104A192)
9. Letter from U.S. Nuclear Regulatory Commission to Entergy Nuclear Operations, Inc., "Vermont Yankee Nuclear Power Station - Issuance of Amendment for Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition (CAC No. MF3714)," dated October 7, 2015 (ML15117A551)
10. Letter from U.S. Nuclear Regulatory Commission to Dominion Energy Kewaunee, Inc., "Kewaunee Power Station - Issuance of Amendment for Permanently Shutdown and Defueled Technical Specifications and Certain License Conditions (TAC No. MF1952)," dated February 13, 2015 (ML14237A045)
11. Letter from U.S. Nuclear Regulatory Commission to Southern California Edison Company, "San Onofre Nuclear Generating Station, Units 2 and 3 – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC Nos. MF3774 and MF3775)," dated July 17, 2015 (ADAMS Accession No. ML15139A390)
12. Letter from U.S. Nuclear Regulatory Commission to Crystal River Nuclear Plant, "Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC No MF3089)," dated September 4, 2015 (ADAMS Accession No. ML15224B286)
13. NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," Draft for Comment March 1998 (ML082330233)
14. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000, (ADAMS Accession No. ML003716792)
15. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U.S Nuclear Regulatory Commission, "Response to Generic Letter 2016-01," November 3, 2016 (ML16308A470)
16. NUREG-1801, Revision 2, “Generic Aging Lessons Learned (GALL) Report — Final Report,” December 2010
17. NRC Information Notice 2009-26, "Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool," dated October 28, 2009
18. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U.S Nuclear Regulatory Commission, "Report on Status of Decommissioning Funding for Reactor," dated March 31, 2009 (ML090900463)
19. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "License Amendment Request Regarding to Implement BWRVIP-18, Revision 2A," dated August 30, 2017 (ML17242A211).
20. Oyster Creek Generating Station - NRC License Renewal Follow-Up Inspection Report 0500021 912009006, dated May 18, 2009 (ML091380379)
21. "Assessment of the Oyster Creek 3-D Finite Element Analysis of the Drywell Shell," dated May 12, 2009 (ML091310413)

22. Letter from Pamela B. Cowan (Exelon Generation, LLC) to U.S. Nuclear Regulatory Commission – “Updated Final Safety Analysis Report (UFSAR), Revision 16, Fire Hazards Analysis Report (FHAR), Revision 15, UFSAR and FHAR Reference Drawings,” dated October 23, 2009 (ML110691283)

23. NRC Safety Evaluation by the Office of Nuclear Reactor Regulation, Direct Transfer of Facility Operating Licenses from Amergen Energy Company, LLC to Exelon Generation Company, LLC, License Nos. NPF-62, DPR-16 and DPR-50 Clinton Power Station, Unit No. 1; Oyster Creek Nuclear Generating Station; and Three Mile Island Nuclear Station, Unit 1; Docket Nos. 50-461, 50-219, 72-15 and 50-289, dated December 23, 2008 (ML082750072)

24. Letter from U.S. Nuclear Regulatory Commission to Mr. I. R. Finfrock, "Order for Modification of License Concerning Environmental Qualification of Safety-Related Electrical Equipment (SEP Topic III-12) – Oyster Creek Nuclear Generating Station," dated October 24, 1980 (ML011150584)

Attachment 2

Proposed Technical Specifications (Marked-Up Pages)

Oyster Creek Nuclear Generation Station
Renewed Facility Operating License No. DPR-16
NRC Docket No. 50-219

Changes to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specification (TS)

(49 pages)

Section	Pages
RFOL	1 through 9
TS Table of Contents	i through iii
TS Section 1 Definitions	1.0-1 & 1.0-9
TS Section 3/4.0 Limiting Conditions For Operation and Surveillance Requirements	3/4.0-1 & 3/4.0-2
TS Bases Section 3/4.0	B3/4.0-1, 3.0.3, 3.0.4, B3/4.0-2, B3/4.0-3, B3/4.0-4
TS 3/4.1 Spent Fuel Storage	3/4.1-1
TS Bases 3/4.1	B3/4.1-1
TS 3/4.2 Radioactive Liquid Storage	3/4.2-1
TS Bases 3/4.2	B3/4.2-1
TS Section 5 Design Features	5.1-1, 5.2-1, 5.3-1
TS Bases Section 5	B5.1-1, B5.2-1
TS Section 6.0 Administrative Controls	6-1 through 6-14

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-16, has now found that:

   A. The application for a Renewed Facility Operating License No. DPR-16 filed by the applicant 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 and all required notifications to other agencies or bodies have been duly made;

   B. ***DELETED*** ~~Construction of the Oyster Creek Nuclear Generating Station (Oyster Creek or the facility) has been completed in conformity with Provisional Construction Permit No. CPPR-15; the application, as amended; the provisions of the Act; and the rules and regulations of the Commission.~~

   C. ***DELETED*** ~~Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the term of this Renewed Facility Operating License No. DPR-16 on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;~~

   D. The facility will ***be maintained*** ~~operate~~ in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission ~~(except as exempted from compliance in Section 2.D. below);~~

E. There is reasonable assurance (i) that the activities authorized by this ~~operating~~ license 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 rules and regulations set forth in 10 CFR Chapter I ~~(except as exempted from compliance in Section 2.D. below);~~

F. Exelon Generation Company, LLC (Exelon Generation Company) is technically qualified to engage in the activities authorized by this license in accordance with the rules and regulations of the Commission;

G. Exelon Generation Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

I. The receipt, possession and use of source, byproduct, and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70; and

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

2. Facility Operating License No. DPR-16, dated July 2, 1991, as amended, is superseded in its entirety by Renewed Facility Operating License No. DPR-16, hereby issued to Exelon Generation Company, to read as follows:

A. This renewed license applies to the Oyster Creek Nuclear Generating Station, a boiling-water reactor and associated equipment (the facility). The facility is located in Ocean County, New Jersey, and is described in the licensee's Updated Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Exelon Generation Company:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, to possess~~,~~ ***and*** use~~, and operate~~ Oyster Creek Nuclear Generation Station at the designated location on the Oyster Creek site in Ocean County, New Jersey, in accordance with the procedures and limitations set forth in this renewed license;

(2) Pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material ***that was used*** as reactor fuel, in accordance with the
limitations for storage ~~and amounts required for reactor operation~~, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(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 ***that were used*** for reactor startup, sealed sources ***that were used*** for ***calibration of*** reactor instrumentation and ***are used in*** 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~~ ***that were*** 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) ***DELETED***~~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. ***[###]***, are hereby ~~incorporated in the license~~***replaced with the Permanently Defueled Technical Specifications (PDTS)***. Exelon Generation Company shall ~~operate~~***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specifications.

(3) ***DELETED***~~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.~~

(4) Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Oyster Creek Nuclear Generating Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 5." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by Renewed License Amendment No. 280 and modified by License Amendment No***s***. 288 ***and [###]***.

(5) ***DELETED***~~Inspections of core spray spargers, piping and associated components will be performed in accordance with BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," as approved by NRC staffs Final Safety Evaluation Report dated December 2, 1999.~~

(6) ***DELETED***~~Long Range Planning Program – Deleted~~

(7) ***DELETED***~~Reactor Vessel Integrated Surveillance Program~~

~~Exelon Generation Company is authorized to revise the Updated Final Safety Analysis Report (UFSAR) to allow implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel Integrated Surveillance Program as the basis for demonstrating compliance with the requirements of Appendix H to Title 10 of the *Code of Federal Regulations* Part 50, "Reactor Vessel Material Surveillance Program Requirements," as set forth in the licensee's application dated December 20, 2002, and as supplemented on May 30, September 10, and November 3, 2003.~~

~~All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program appropriate for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.~~

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

(8) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(9) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(10) ***DELETED***~~Upon implementation of Amendment No. 265 adopting TSTF-448, Revision 3, the assessment of CRE habitability as required by Specification 6.22.c.(ii), and the measurement of CRE pressure as required by Specification 6.22.d, shall be considered met. Following implementation:~~

~~(a) The first performance of the periodic assessment of CRE habitability, Specification 6.22.c.(ii), shall be within 3 years, plus the 9-month allowance of Specification 1.24.~~

~~(b) The first performance of the periodic measurement of CRE pressure, Specification 6.22.d, shall be within 24 months, plus the 180 days allowed by Specification 1.24, as measured from the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.~~

(11) ***DELETED***~~Inspection of Drywell Sand Bed Region~~

~~The licensee shall perform full scope inspections (as defined in Appendix A of the license renewal safety evaluation report dated March 20, 2007, and summarized in the Updated Final Safety Analysis Report (UFSAR)) of the drywell sand bed region every other refueling outage beginning in the refueling outage prior to April 9, 2009.~~

(12) ***DELETED***~~Drywell Trenches~~

~~The licensee shall monitor the drywell trenches (as defined in Appendix A of the license renewal safety evaluation report dated March 20, 2007, and summarized in the UFSAR) every refueling outage to identify and eliminate the sources of water and shall receive NRC approval prior to restoring the trenches to their original design configuration.~~

(13) ***DELETED***~~Engineering Study of Refueling Cavity Liner~~

~~The licensee shall perform an engineering study prior to April 9, 2009 to identify options to eliminate or reduce the leakage in the facility cavity liner.~~

(14) ***DELETED***~~Three Dimensional Finite-Element Analysis of Drywell Shell~~

~~The licensee shall perform a three dimensional finite-element analysis of the drywell shell and shall provide to the NRC staff a report of the results prior to April 9, 2009.~~

(15) ***DELETED***~~UFSAR Supplement Changes~~

~~The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4), as modified by an exemption granted by letter dated July 7, 2004 (ADAMS Accession No. ML041340673), following the issuance of this renewed operating license. Until that update is complete, Exelon Generation Company may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Exelon Generation Company evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.~~

(16) License Renewal Commitments

The UFSAR supplement, as revised, describes certain future activities to be completed prior to April 9, 2009, and during the term of this renewed operating license No. DPR-16. Exelon Generation Company shall complete these activities in accordance with Appendix A of NUREG-1875, “Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station,” dated March 2007, as supplemented on September 19, 2008, and shall notify the NRC in writing when implementation of those activities required prior to April 9, 2009 are complete and can be verified by NRC inspection.

(17) Biological Opinion

Within 30 days from the issuance date of the renewed license, Exelon Generation Company shall comply with the terms and conditions of the Incidental Take Statement associated with certain sea turtles in the Biological Opinion in effect or as subsequently issued by the National Marine Fisheries Service regarding operation of the facility.

D. ***DELETED***~~The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."~~

~~This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to the licensee in letters dated March 24, 1986 and June 25,1990.~~

~~The facility has also been granted certain exemptions from the requirements of Section III.J of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to emergency lighting that shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto. This exemption was granted and sent to the licensee in a letter dated February 12, 1990.~~

~~In addition, the facility has been granted certain exemptions from Section 55.45(b)(2)(iii) and (iv) of 10 CFR Part 55, "Operators' Licenses." These sections contain requirements related to site-specific simulator certification and require that operating tests will not be administered on other than a certified or an approved simulation facility after May 26, 1991. These exemptions were granted and sent to the licensee in a letter dated March 25, 1991.~~

~~These exemptions granted pursuant to 10 CFR 50.12 are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.~~

E. ***DELETED***~~Deleted~~

F. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

3. Sale and License Transfer Conditions:

A. ***DELETED***~~Deleted.~~

B. ***DELETED***~~Deleted.~~

C. ***DELETED***~~Deleted.~~

D. ***DELETED***~~Deleted~~

E. ***DELETED***~~Deleted~~

E. ***DELETED***~~Deleted~~

F. ***DELETED***~~Deleted~~

G. ***DELETED***~~Deleted~~

H. ***DELETED***~~Deleted~~

I. ***DELETED***~~Deleted~~

J. ***DELETED***~~Deleted~~

K. ***DELETED***~~Deleted~~

L. DELETED

M. ***DELETED***~~At the time of the closing of the transfer of Oyster Creek, and the respective license from AmerGen Energy Company, LLC (AmerGen) to Exelon Generation Company, AmerGen shall transfer to Exelon Generation Company ownership and control of AmerGen Oyster Creek NQF, LLC, and AmerGen Consolidation, LLC shall be merged into Exelon Generation Consolidation, LLC. Also at the time of the closing, decommissioning funding assurance provided by Exelon Generation Company, using an additional method allowed under 10 CFR 50.75 if necessary, must be equal to or greater than the minimum amount calculated on that date pursuant to, and required by 10 CFR 50.75 for Oyster Creek. Furthermore, funds dedicated for Oyster Creek prior to closing shall remain dedicated to Oyster Creek following the closing. The name of AmerGen Oyster Creek NQF, LLC shall be changed to Exelon Generation Oyster Creek NQF, LLC at the time of the closing.~~

4. This license is effective as of the date of issuance and ~~shall expire at midnight on April 9, 2029~~***is effective until the Commission notifies the licensee in writing that the license is terminated.***

FOR THE NUCLEAR REGULATORY COMMISSION

Bruce S. Mallett **/RA/**
Deputy Executive Director for Reactor
and Preparedness Programs
Office of the Executive Director for Operations

Attachment:
Appendices A and B -
Technical Specifications

Date of Issuance: April 8, 2009

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~~*Issued by NRC Order dated 10-24-80~~

SECTION I
DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of the specifications.

1.1 ~~OPERABLE-OPERABILITY~~ ***ACTIONS***

ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.

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

~~A verification of operability is an administrative check, by examination of appropriate plant records (logs, surveillance test records) to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine operability.~~

1.2 ~~OPERATING~~ ***CERTIFIED FUEL HANDLER***

~~Operating means that a system or component is performing its required function.~~

1.3 ~~POWER OPERATION~~ ***NON-CERTIFIED OPERATOR***

~~Power operation is any operation when the reactor is in the startup mode or run mode except when primary containment integrity is not required.~~

~~1.4 STARTUP MODE~~

~~The reactor is in the startup mode when the reactor mode switch is in the startup mode position. In this mode, the reactor protection system scram trips initiated by condenser low vacuum and main steam line isolation valve closure are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure is bypassed; the IRM trips for rod block and scram are operable; and the SRM trips for rod block are operable.~~

~~1.5 RUN MODE~~

~~The reactor is in the run mode when the reactor mode switch is in the run mode position. In this mode, the reactor protection system is energized with APRM protection and the control rod withdrawal interlocks are in service.~~

~~1.6 SHUTDOWN CONDITION~~

~~The reactor is in the SHUTDOWN CONDITION when there is fuel in the reactor vessel, the reactor is subcritical, all operable control rods are fully inserted, and the mode switch is in the shutdown mode position. In this position, a control rod block is initiated.~~

~~1.49 RATED THERMAL POWER (RTP)~~

~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.~~

~~1.50 THERMAL POWER~~

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

~~1.51 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.23.~~

1.~~5~~2 CERTIFIED FUEL HANDLER

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.~~5~~3 NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

SECTION 3/***4***

LIMITING CONDITIONS FOR OPERATION ***AND SURVEILLANCE REQUIREMENTS***

3/***4***.0 LIMITING CONDITIONS FOR OPERATION ~~(GENERAL)~~ ***AND SURVEILLANCE REQUIREMENT APPLICABILITY***

Applicability: Applies to all Limiting Conditions for Operation ***and Surveillance Requirements.***

Objective: To preserve the single failure criterion for safety systems.

LCO Applicability:

LCO 3.0.1 ***LCOs shall be met during the specified conditions in the TS, except as provided in TS 3.0.2.***

LCO 3.0.2 ***Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.***

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

~~A. In the event Limiting Conditions for Operation (LCOs) and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable. Exceptions to the requirements shall be stated in the individual specifications.~~

~~B. When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of applicable LCOs., provided (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfieid, the unit shall be placed in COLD SHUTDOWN within the following 30 hours or within the time specified in the applicable specification. This specification is not applicable in COLD SHUTDOWN or the REFUEL MODE.~~

~~C. When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:~~

~~1. When the associated LCO requirement permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or~~

~~2. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the applicability, and the establishment of risk management actions, if appropriate; exceptions to this specification are stated in the individual Specifications, or~~

~~3. When an allowance is stated in the individual value, parameter, or other Specification.~~

~~This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.~~

~~Section 4 Surveillance Requirements~~

~~4.0~~ Surveillance Requirement Applicability

SR 4.0.1 Surveillance requirements shall be met during the ~~modes or other~~ specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in 4.0.2. Surveillances do not have to be performed on ~~inoperable equipment or~~ variables outside specified limits.

SR 4.0.2 If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

SR 4.0.3 Entry into a~~n OPERATIONAL CONDITION or other~~ specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillance ~~have~~ has been met within ~~their~~ its specified frequency, except as provided by 4.0.2. ~~When an LCO is not met due to surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.C.~~

This provision shall not prevent entry into ~~OPERATIONAL CONDITIONS or~~ other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.

SR 4.0.4 The specified frequency for each SR is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.

~~BASES:~~ ~~Surveillance Requirement 4.0.1 establishes the requirement that surveillance requirements must be met during the modes or other specified conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual surveillance requirements. This specification is to ensure that surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a surveillance within the specified frequency constitutes a failure to meet an LCO.~~

B3/4.0 BASES FOR LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENT APPLIACIABILTY

~~Bases:~~

LCO 3.0.1 through LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 establishes the applicability statement within each individual specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the applicability statement of each Specification).

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated actions shall be met. The completion time of each required action for an ACTIONS condition is applicable from the point in time that an actions condition is entered. The required actions establish those remedial measures that must be taken within specified completion times when the requirements of an LCO are not met. This specification establishes that:

a. ***Completion of the required actions within the specified completion times constitutes compliance with a specification; and***

b. ***Completion of the required actions is not required when an LCO is met within the specified completion time, unless otherwise specified.***

Completing the required actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual specifications.
~~Specification 3.0.A delineates the action to be taken for circumstances not directly provided for in the system LCOs and whose occurrence would violate the intent of the specification.~~

~~Specification 3.0.B delineates what additional conditions must be satisfied to permit operation to continue, consistent with the specifications for power sources, when a normal or emergency power source~~
~~is not operable. It allows operation to be governed by the time limits of the specifications associated with the LCOs for the normal or emergency power source, not the individual specifications for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source. In addition, it specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a safety~~
~~subsystem, train, component, or device in another division is inoperable for another reason.~~

~~Specification 3.0.C establishes limitations on changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability when an LCO is not met. It allows placing the plant in an OPERATIONAL CONDITION or other specified condition stated in the Applicability (e.g., the Applicability desired to be entered) when plant conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.C.1, LCO 3.0.C.2, or LCO 3.0.C.3.~~

~~LCO 3.0.C.1 allows entry into an OPERATION CONDITON or other specified condition in the Applicability with the requirements of the LCO met that permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time. Compliance with LCO conditions that permit continued operation of the unit for an unlimited period of time in an OPERATIONAL CONDITON or other specifed condition in the Applicability provides an acceptable level of safety for continued operation. This is without regard for the status of the unit before or after the change in OPERATIONAL CONDITON. Therefore, in such cases, entry into an OPERATION CONDITION or other specified condition in the Applicability may be made in accordance with the provisions of the LCO conditions.~~

~~LCO 3.0.C.2 allows entry into an OPERATIONAL CONDITION or other specified condition in the Applicability with the requirements of the LCO met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and~~

~~establishment of risk management actions, if appropriate.~~

~~The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.C.2, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plant." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for the conduct of the risk assesment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed change in OPERATIONAL CONDITION is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the Limiting Condition for Operation would be met prior to the expiration of the specified allowable time interval requiring action to exit the LCO.~~

~~LCO 3.0.C.2 may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailabity of multiple systems and components.~~

~~The results of the risk assessment shall be considered in determining the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and any corresponding risk management actions. The Specification 3.0.C.2 risk assessments do not have to be documented.~~

~~The Technical Specifications allow continues operation with equipment unavailable in the RUN MODE for the duration of the specified time interval. Since this allowable, and since in general the risk impact in that particular OPERATIONAL CONDITION bounds the risk of transitioning into and through the applicable OPERATIONAL CONDITIONS or other specified conditions in the Applicability of the Limiting Condition for Operation, the use of the Specification 3.0.C.2 allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of system and components that have been determined to be more important to risk and use of the Specification 3.0.C.2 allowance is prohibited. The Limiting Condition for Operations governing these systems and components contain Notes prohibiting the use of Specification 3.0.C.2 by stating that Specification 3.0.C.2 is not applicable.~~

~~Specification 3.0.C.3 allows entry in an OPERATIONAL CONDITION or other specified condition in the Applicability with the Limiting Condition for Operation not met based on a Note in the Specification which states Specification 3.0.C.3 is applicable. These specific allowances permit entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability when the associated LCO requirements do not provide for continued operation for an unlimited period to time and a risk assessment has not been performed. This allowance may apply to all the LCO conditions or to a specific requirement of a Specification. The risk assessments performed to justify the use of Specification 3.0.C.2 usually only consider systems and components. For this reason, Specification 3.0.C.3 is typically applied to Specifications which describe values and parameters, and may be applied to other Specifications based on NRC plant-specific approval.~~

~~The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated OPERATIONAL CONDITION or other specified condition in the Applicability.~~

~~The provisions of Specification 3.0.C shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with LCO requirements. In addition, the provisions of Specification 3.0.C shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in OPERATIONAL CONDITION or other specified condition in the Applicability associated with transitioning from POWER OPERATION to STARTUP MODE, STARTUP MODE to SHUTDOWN CONDITION, and SHUTDOWN CONDITION to COLD SHUTDOWN CONDITION.~~

~~Upon entry into an OPERATIONAL CONDITION or other specified condition in the Applicability with the Limiting Condition for Operation not met, Specification 3.0.A requires entry into the LCO requirements until the Condition is resolved, until the Limiting Condition for Operation is met, or until the unit is not within the Applicability of the Technical Specification.~~

~~Surveillances do not have to be performed on the associated inoperable (or on variables outside the specified limits), as permitted by Specification 4.0.1. Therefore, utilizing Specification 3.0.C is not a violation of Specification 4.0.1 or Specification 4.0.3 for any Surveillances that have not been performed on inoperable equipment. However, Surveillance Requirements must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected Limiting Condition of Operation.~~

BASES: ~~Surveillance Requirement~~***SR*** 4.0.1 establishes the requirement that surveillance requirements must be met during the ~~modes or other~~ specified conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual surveillance requirements. This specification is to ensure that surveillances are performed to verify ~~the OPERABILITY of systems and components, and~~ that variables are within specified limits. Failure to meet a surveillance within the specified frequency constitutes a failure to meet an LCO.

~~Systems and components are assumed to be OPERABLE when the associated surveillance requirements have been met. Nothing in this specification, however, is to be construed as implying that systems or components are OPERABLE when:~~

~~a. The systems or components are known to be inoperable, although still meeting the surveillance requirements; or~~

~~b. The requirements of the surveillance(s) are known to be not met between required surveillance performances.~~

~~Surveillances do not have to be performed when the unit is in a mode or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.~~

~~Surveillances, including surveillances invoked by required actions, do not have to be performed on inoperable equipment because the actions define the remedial measures that apply. Surveillances have to be met and performed prior to returning equipment to OPERABLE status.~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable surveillances are not failed. Post maintenance testing may not be possible in the current mode or other specified conditions in the applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a mode or other specified condition where other necessary post maintenance tests can be completed.~~

~~Surveillance Requirement~~***SR*** 4.0.2 establishes the flexibility to defer declaring ~~affected equipment inoperable or~~ an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the surveillance has not been performed in accordance with ~~surveillance requirement~~***SR*** 4.0.***4***~~2~~, and not at the time that the specified frequency was not met.

This delay period provides adequate time to complete surveillances that have been missed. This delay period permits the completion of a surveillance before complying with required actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of ~~unit~~ ***facility*** conditions, adequate planning, availability of personnel, the time required to perform the

surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

~~When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operational situations, or requirements of regulations (e.g., prior to entering power operation after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Surveillance Requirement 4.0.2 allows the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity. Surveillance requirement 4.0.2 provides a time limit for, and allowances for the performance of, surveillance that become applicable as a consequence of mode changes imposed by required actions.~~

Failure to comply with specified surveillance frequencies is expected to be an infrequent occurrence. Use of the delay period established by ~~Surveillance Requirement~~***SR*** 4.0.***4***~~2~~ is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required ~~or shutting the plant down~~ to perform the surveillance) and impact on any analysis assumptions, in addition to ~~unit~~ ***facility*** conditions, planning, availability of personnel, and the time required to perform the surveillance. ~~This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should cmmensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action.~~ All missed surveillances will be placed in the licensee's Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the ~~equipment is considered inoperable or the~~ variable is considered outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then ~~the equipment is inoperable, or~~ the variable is outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon the failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the completion time of the actions, restores compliance with Surveillance Requirement 4.0.1.

~~Specification~~ ***SR*** 4.0.3 establishes the requirement that all applicable ~~Surveillance Requirement (~~SRs~~)~~ must be met before entry into a~~n OPERATIONAL CONDITION or other~~ specified condition in the Applicability.

This Specification ensures that system ~~and component OPERABILITY requirements and~~ variable limits are met before entry into ~~OPERATIONAL CONDITIONS or other~~ specified conditions in the Applicability for which these ~~systems and components~~***variable limits*** ensure safe operation of the ~~unit~~***facility***. ~~The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering and associated OPERATIONAL CONDITION or other specified condition in the Applicability.~~

~~A provision is included to allow entry into an OPERATIONAL CONDITION or other specified condition in the Applicability when a Limiting Condition for Operation is not met due to a Surveillance not being met in accordance with Specification 3.0.C.~~

~~However, in certain circumstances, failing to meet a Surveillance Requirement will not result in Specification 4.0.3 restricting an OPERATIONAL CONDITION change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per Specification 4.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, Specification 4.0.3 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Surveillance time interval does not result in a Specification 4.0.3 restriction to changing OPERATIONAL CONDTIONS or other specified conditions of the Applicability. However, since the Limiting Condition for Operation is not met in this instance, Specification 3.0.C will govern any restrictions that may (or may not) apply to OPERATIONAL CONDITION or other specified condition changes. Specification 4.0.3 does not restrict changing OPERATIONAL CONDITIONS or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Surveillance time interval, provided the requirement to declare the Limiting Condition for Operation not met has been delayed in accordance with Specification 4.0.2.~~

~~The provisions os Specification 4.0.3 shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with LCO requirements. In addition, the provision of Specification 4.0.3 shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in OPERATIONAL CONDITION or other specified condition in the Applicability associated with transitioning from POWER OPERATION to STARTUP MODE, STARTUP MODE to SHUTDOWN CONDITION, and SHUTDOWN CONDITION to COLD SHUTDOWN CONDITION.~~

SR 4.0.4 establishes the requirements for meeting the specified frequency for surveillances. SR 4.0.4 permits a 25% extension of the interval specified in the frequency. This extension facilitates surveillance scheduling and considers facility conditions that may not be suitable for conducting the surveillance (e.g., transient conditions or other ongoing surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the surveillance at its specified frequency. This is based on the recognition

that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs.

The provisions of SR 4.0.4 are not intended to be used repeatedly merely as an operational convenience to extend surveillance intervals (other than those consistent with refueling intervals).

3/4.1 SPENT FUEL STORAGE

***Applicability**: During movement of irradiated fuel assemblies in the spent fuel pool.*

***Objective**: To assure safe storage of spent fuel.*

***LCO**: 3.1 Spent Fuel Pool Water Level*

Whenever irradiated fuel is stored in the spent fuel storage pool, water level shall be maintained at a level ≥ 117 feet 8 inches (elevation above sea level) with the exception of planned cask movements.

***ACTIONS**:*

Condition	Required Action	Completion Time
Spent fuel pool water level is not within limit.	Suspend movement of irradiated fuel assemblies and movement of loads over the storage racks containing fuel.	Immediately

SURVEILLANCE REQUIREMENTS

Surveillance		Frequency
4.1	Verify the spent fuel pool water level is ≥ 117 feet 8 inches.	24 hours

Basis: *LCO 3.1, "Spent Fuel Pool Water Level," specifies requirements to ensure that the minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident (FHA) in the spent fuel pool (SFP). The water also provides shielding during the movement of spent fuel.*

The required minimum water level in the SFP meets the assumptions of the FHA described in calculation C-1302-226-E310-460 and Chapter 15.7.4 of the UFSAR. The resultant dose limits at the exclusion area boundary are within the critieria of RG1.183.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2. The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.4.

FHA is evaluated for dropping an irradiated fuel assembly onto irradited fuel bundles stored in the SFP. The consequences of a FHA in the SFP are documented in FSAR Chapter 15. The water level in the SFP provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the building atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a FHA.

The SFP water level is monitored in terms of elevation above mean sea level. Elevation 117 feet 8 inches corresponds to the SFP low level alarm in the Control Room. Since the pool has no installed drains, level cannot be lowered by the cooling system below the level of the weirs. At the normal 400 gpm flow rate, the pool level is about three inches above the weir level, and the overflow just equals the 400 gpm being supplied to the pool from the diffusers. At the SPF low level alarm level, the pool contains a depth of approximately 37 feet of water (approximately 23 feet above active fuel), providing adequate shielding for normal building occupancy by operating personnel.

LCO 3.1 requires that when the water level in the SFP is lower than the required level, the movement of irradiated fuel assemblies in the SFP is to be "immediately" suspended. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, such that the suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring in the SFP when the level is below the required elevation. This specification is not meant to affect spent fuel cask movements during planned SFP level adjustments. The FSAR Chapter 15 analysis states that a spent fuel cask drop accident is no longer credible since the reactor building crane has been upgraded to be single-failure proof.

Surveillance Requirement (SR) 4.1 verifies that sufficient SFP water is available in the event of a fuel handling accident. The water level in the SFP must be checked periodically. The frequency of every 24 hours is acceptable based on operating experience, considering that the water volume in the pool is normally stable and water level changes are controlled by unit procedures.

The fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.2 RADIOACTIVE LIQUID STORAGE

Applicability: Applies at all times to ~~specified~~ outdoor tanks used to store radioactive liquids.

Objective: To assure that radioactive ~~material is~~ ***effluents are*** not released to the environment in an uncontrolled manner and to assure that the radioactive concentrations of any material released is kept as low as is reasonably achievable and, in any event, within the limits of 10 CFR ***P***art 20.1301 and 40 CFR Part 190.10(a).

LCO: 3.2 The quantity of radioactive material, excluding tritium, noble gases, and radionuclides having half-lives shorter than three days, contained in ~~any of the following~~ outdoor ***storage*** tanks shall not exceed 10.0 curies~~:~~***. Included in this specification are all outdoor storage tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.***

~~a. Waste Surge Tank, HP-T-3~~

~~b. Condensate Storage Tank~~

ACTIONS:

Condition	***Required Action***	***Completion Time***
In the event the quantity of radioactive material in any ~~of the~~ ***applicable storage tanks*** ~~named~~ ***exceeds 10.0 curies.***	Begin treatment and continue it until the total quantity of radioactive material in the tank is 10 curies or less, and describe the reason for exceeding the limit in the next Annual Effluent Release Report.	As soon as reasonably achievable

SURVEILLANCE REQUIREMENTS

Surveillance		***Frequency***
4.2	Liquids contained in ~~the following~~ ***outdoor storage*** tanks ***included in this specification*** 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.~~	***Once per 7 days when radioactive liquid is being added to the tank.***

Basis: LCO 3.2, "Radioactive Liquid Storage:"

Restricting the quantity of radioactive material contained in the defined outdoor storage tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2 in the canal at the Route 9 bridge.

The specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

SECTION 5

DESIGN FEATURES

5.1 SITE

A. The reactor (center line) is located 1,358 feet west of the east boundary of New Jersey State Highway Route 9 which is the minimum exclusion distance as defined in 10 CFR 100.3. The licensee will at all times retain the complete authority to determine and maintain sufficient control of all activities through ownership, easement, contract and/or other legal instruments on property which is closer to the reactor (center line) than 1,358 feet. This includes the authority to exclude or remove personnel and property within the minimum exclusion distance.

~~B. The reactor building, standby gas treatment system and stack shall comprise a secondary containment in such fashion to enclose the primary containment in order to provide for controlled elevated release of the reactor building atmosphere under accident conditions.~~

~~5.2 CONTAINMENT~~

~~A. The primary containment shall be of the pressure suppression type having a drywell and an absorption chamber constructed of steel. The drywell shall have a volume of approximately 180,000 ft^3 and conforms to ASME Boiler and Pressure Vessel Code, Section VIII, for an internal pressure of 44 psig at 292°F and an external pressure of 2 psig at 150°F to 205°F. The absorption chamber shall have a total volume of approximately 210,000 ft^3. It is designed to conform to ASME Boiler and Pressure Vessel Code, Section VIII, for an internal pressure of 35 psig at 150°F and an external pressure of 1 psig at 150°F.~~

~~B. Penetrations added to the primary containment shall be designed in accordance with standards set forth in Section V-1.5 of the Facility Description and Safety Analysis Report. Piping passing through such penetrations shall have isolation valves in accordance with standards set forth in Section V-1.6 of the Facility Description and Safety Analysis Report.~~

~~BASIS~~

~~The drywell pressure of 44 psig is based upon a conservatively calculated peak drywell pressure of 38.1 psig plus an added 15% allowance. The calculated peak pressure results from a design basis loss of coolant accident (DBLOCA). The corresponding coincident drywell temperature of 292°F is the saturated steam temperature of the containment atmosphere for the 44 psig pressure. The specified coincident pressure and temperature condition represent the bounding case for the structural pressure/temperature design of the drywell.~~

5.~~3~~***2*** ~~AUXILIARY EQUIPMENT~~***SPENT FUEL STORAGE***

5.~~3~~***2***.1 ***Spent*** Fuel Storage

A. The ***spent*** fuel storage facilities are designed and shall be maintained with a K-effective equivalent to less than or equal to 0.95 including all calculational uncertainties.

~~B. Deleted~~

~~C. Deleted~~

~~D~~***B***. The temperature of the water in the spent fuel storage pool, measured at or near the surface, shall not exceed 125°F.

~~E~~***C***. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 3035.

B5 Bases:

B5.1 - Site

Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the hig~~i~~hway, railroad, or waterway, in case of emergency, to protect the public health and ~~a~~safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operaiton of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazard to the public health and safety will result.

Activities unrelated to plant operation within the exclusion area are acceptable provided:

(a) Such activities, including accidents associated with such activities, represent no hazard to the plant or have been shown to be accommodated as part of the plant design basis.

(b) The licensee is aware of such activities and has made appropriate arrangments to evacuate persons engaged in such activities, in the event of an accident, and

(c) There is reasonable assurance that persons engaged in such activities can be evacuated without receiving radiation doses in excess of the guideline values given in 10 CFR Part 100.

Contract provisions for property agreements in the exclusion area must ensure that the licensee retains sufficient control of all activities in the exclusion area including the authority to exclude or removal personnel and property, thereby, (1) maintaining compliance with 10 CFR Part 100 radiological limits for the exclusion area, including evacuation when necessary, and (2) ensuring that any activities, now or in the future, in the exclusion area would not negatively effect nuclear safety, safe plant operations or violate current plant design or licensing bases.

Any property transactions in the "exclusion area", as is the case for any activity which has the potential to adversely affect nuclear safety or safe plant operations, requires a specific safety evaluation and 50.59 review.

~~References:~~

~~(1) 10 CFR Part 100, "Reactor Site Criteria"~~
~~(2) NRC Standard Review Plan, NUREG-0800 (Formerly NUREG-75/087), Chapter 2.1.2, "Exclusion Area Authority and Control", Rev. 2, July 1981.~~

B5.2 – Spent Fuel Storage

The specification of a K-effective less than or equal to 0.95 in fuel storage facilities assures an ample margin from criticality. This limit applies to unirradiated fuel in both the dry storage vault and the spent fuel racks as well as irradiated fuel in the spent fuel racks. Criticality analyses were performed on the poison racks to ensure that a K-effective of 0.95 would not be exceeded. The analyses took credit for burnable poisons in the fuel and included manufacturing tolerances and uncertainties as described in Section 9.1 of the FSAR. Calculational uncertainties described in 5.~~3~~2.1.A are explicitly defined in FSAR Section 9.1.2.3.9. Any fuel stored in the fuel storage facilities shall be bounded by the analyses in these reference documents.

The effects of a dropped fuel bundle onto stored fuel in the spent fuel storage facility has been analyzed. This analysis shows that the fuel bundle drop would not cause doses resulting from ruptured fuel pins that exceed 10 CFR 100 limits ~~(1,2,3)~~.

Detailed structural analysis of the spent fuel pool was performed using loads resulting from the dead weight of the structural elements, the building loads, hydrostatic loads from the pool water, the weight of fuel and racks stored in the pool, seismic loads, and loads due to thermal gradients in the pool floor and the walls. Thermal gradients result in two loading conditions: normal operating and the accident conditions with the loss of spent fuel pool cooling. For the normal condition, the reactor building air temperature was assumed to vary between 65°F and 110°F while the pool water temperature varied between 85°F and 125°F. The most severe loading from the normal operating thermal gradient results with reactor building air temperatures at 65°F and the water temperature at 125°F. Air temperature measurements made during all phases of plant operation in the shutdown heat exchanger room, which is directly beneath part of the spent fuel pool floor slab, show that 65°F is the appropriate minimum air temperature. The spent fuel pool water temperature will alarm control room before the water temperature reaches 120°F.

Results of the structural analysis show that the pool structure is structurally adequate for the loadings associated with the normal operation and postulated accidents ~~(5) (6)~~. The floor framing was also found to be capable of withstanding the steady state thermal gradient conditions with the pool water temperature at 150°F without exceeding ACI Code requirements. The walls are also capable of operation at a steady state condition with the pool water temperature at 140°F ~~(5)~~.

Since the cooled fuel pool water returns at the bottom of the pool and the heated water is removed from
the surface, the average of the surface temperature and the fuel pool cooling return water is an appropriate estimate of the average bulk temperature; alternately the pool surface temperature could be conservatively used.

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during the Plant Manager's absence.

The Plant Manager or the designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect safe storage and maintenance of spent nuclear fuel.

6.1.2 The Shift Manager shall be responsible for the shift command function.

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safe storage and handling of spent nuclear fuel.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including facility organization positions. These relationships shall be documented and updated as appropriate, in the form of organizational descriptions. These organizational descriptions will be documented in the Updated FSAR and updated in accordance with 10 CFR 50.71e.

b. The Plant Manager shall be responsible for overall facility safe operation and shall have control over those onsite activities necessary for safe storage and maintenance of spent nuclear fuel.

c. A responsible officer shall have corporate responsibility for the safe storage and handling of spent nuclear fuel and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure safe management of spent nuclear fuel.

d. Individuals who train the CERTIFIED FUEL HANDLERs and those who carry out the health physics and quality assurance functions may report to the appropriate manager on site; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

6.2.2 Facility Staff

The facility organization shall meet the following:

a. Each on duty shift shall include at least the following shift staffing:

- One (1) Shift Manager (see f. below)
- One (1) NON-CERTIFIED OPERATOR (see g. below)

b. Shift crew composition may be one less than the minimum requirements of 6.2.2.a for a period of time not to exceed two hours, in order to accommodate unexpected absence of on-duty shift crew members. Immediate action must be

taken to restore the shift crew composition to within the requirements given above. During such absences, no fuel movement or movement of loads over the spent fuel shall be permitted. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crew member being late or absent.

c. At all times when nuclear fuel is stored in the spent fuel pool, at least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room.

d. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

e. An individual qualified in radiation protection measures shall be on site during movement of fuel and during the movement of loads over the fuel.

f. The Shift Manager shall be a CERTIFIED FUEL HANDLER.

g. The position of NON-CERTIFIED OPERATOR may be filled by a CERTIFIED FUEL HANDLER.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1 of 1978 for comparable positions unless otherwise noted in the Technical Specifications. Technicians and maintenance personnel who do not meet ANSI/ANS 3.1 of 1978, Section 4.5, are permitted to perform work for which qualification has been demonstrated.

6.3.2 The management position responsible for radiological controls shall meet or exceed the qualifications of Regulatory Guide 1.8 (Rev. 1-R, 9/75). Each other member of the radiation protection organization for which there is a comparable position described in ANSI N18.1-1971 shall meet or exceed the minimum qualifications specified therein, or in the case of radiation protection technicians, they shall have at least one year's continuous experience in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations and shall have been certified by the management position responsible for radiological controls as qualified to perform assigned functions. This certification must be based on an NRC approved, documented program consisting of classroom training with appropriate examinations and documented positive findings by responsible supervision that the individual has demonstrated his ability to perform each specified procedure and assigned function with an understanding of its basis and purpose.

6.3.3 The NRC approved training and retraining program for CERTIFIED FUEL HANDLERs shall be maintained.

6.4 DELETED

6.5 DELETED

6.6 DELETED

6.7 DELETED

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the items referenced below:

a. The procedures applicable to safe storage of nuclear fuel recommended in Appendix "A" of Regulatory Guide 1.33 as referenced in the ~~QATR~~***Decommissioning Quality Assurance Program (DQAP)***.

b. Surveillance and test activities of equipment that affects nuclear safety and radioactive waste management equipment.

c. Fuel Handling Operations.

d. Security Plan Implementation.

e. Fire Protection Program Implementation.

f. Emergency Plan Implementation.

g. Process Control Plan Implementation.

h. Offsite Dose Calculation Manual Implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1.

6.8.2 Each procedure required by 6.8.1 above, and substantive changes thereto, shall be reviewed and approved prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1, above, may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the licensee's management staff knowledgeable in the area affected by the procedure. For changes which may affect the operational status of facility systems or equipment, at least one of these individuals shall be a member of operations management or supervision who is a CERTIFIED FUEL HANDLER.

c. The change is documented, reviewed and approved within 14 days of implementation.

6.8.4 The following programs shall be established, implemented and maintained:

a. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluent and for maintaining the doses to ~~MEMBERS OF THE PUBLIC~~ ***members of the public*** from radioactive effluent as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including the surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2. Limitations on the concentrations of radioactive material released in liquid effluent to the ~~UNRESTRICTED AREA~~***unrestricted area*** conforming to less than the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluent in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM.

4. Limitations on the annual and quarterly doses and dose commitment to a ~~MEMBER OF THE PUBLIC~~ ***members of the public*** from radioactive materials in liquid effluent released to the ~~UNRESTRICTED AREA~~***unrestricted area*** conforming to Appendix I of 10 CFR 50,

5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

6. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in the 31 day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50,

7. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to the ~~UNRESTRICTED AREA~~***unrestricted area*** shall be limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mRems/yr to the total body and less than or equal to a dose rate of 3000 mRems/yr to the skin, and

   b. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mRems/yr to any organ.

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to the ~~UNRESTRICTED AREA~~***unrestricted area*** conforming to Appendix I of 10 CFR 50,

9. Limitations on the annual and quarterly doses to a ~~MEMBER OF THE PUBLIC~~ ***members of the public*** from I-131, I-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent released beyond the ~~SITE BOUNDARY~~***site boundary*** conforming to Appendix I of 10 CFR 50,

10. Limitations on the annual dose or dose commitment to any ~~MEMBER OF THE PUBLIC~~ ***members of the public*** due to releases of radioactivity and to radiation from Uranium fuel cycle sources conforming to 40 CFR Part 190.

b. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

~~2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

~~6.8.5 Station Battery Monitoring and Maintenance Program~~

~~This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications,” of the following:~~

~~a. Actions to restore station battery cells with float voltage < 2.13 volts, and~~

~~b. Actions to equalize and test station battery cells that have been discovered with electrolyte level below the top of the plates.~~

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of 10 CFR, the following identified reports shall be submitted to the Administrator of the NRC Region I office unless otherwise noted.

6.9.1 Routine Reports

a. ~~DELETED~~

~~b. DELETED~~

~~c. DELETED~~

~~d.~~ Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluent and solid waste released from the facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

~~e~~***b***. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in: (1) the ODCM; and, (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

~~Basis: 6.9.1.e - RELOCATED TO THE ODCM.~~

~~f. DELETED~~

~~6.9.2 DELETED~~

~~6.9.3 Unique Reporting Requirements~~

~~Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.~~

~~a. Materials Radiation Surveillance Specimen Reports (4.3A)~~

~~b. (Deleted)~~

~~c. Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.~~

~~d. Deleted~~

~~e-j. Pursuant to the ODCM.~~

~~k. Records of results of analyses required by the Radiological Environmental Monitoring Program.~~

~~l. Failures and challenges to Relief and Safety Valves which do not constitute an LER will be the subject of a special report submitted to the Commission within 60 days of the occurrence. A challenge is defined as any automatic actuation (other than during surveillance or testing) of Safety or Relief Valves.~~

~~m. Plans for compliance with standby liquid control Specifications 3.2.C.3(b) and 3.2.C.3(e)(1) or plans to obtain enrichment test results per Specification 4.2.E.5.~~

~~n. Inoperable high range radioactive noble gas effluent monitor (3.13.H)~~

6.10 RECORD RETENTION

6.10.1 ~~The following records shall be retained for at least five years:~~

~~a. Records and logs of facility operation covering time interval at each power level.~~

~~b. Records and logs of principle maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.~~

~~c. All Licensee Event Reports.~~

~~d. Records of surveillance activities, inspections and calibrations required by these technical specifications.~~

~~e. Records of reactor tests and experiments.~~

~~f. Records of changes made to operating procedures.~~

~~g. Deleted.~~

~~h. Records of sealed source leak tests and results.~~

~~i. Records of annual physical inventory of all source material of record.~~

~~6.10.2 The following records shall be retained for the duration of the Facility Operating License:~~

~~a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.~~

~~b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.~~

~~c. Records of facility radiation and contamination surveys.~~

~~d. Records of doses received by all individuals for whom monitoring was required~~

~~e. Records of gaseous and liquid radioactive material released to the environs.~~

~~f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.~~

~~g. Records of training and qualification for current members of the plant staff.~~

~~h. Records of inservice inspections performed pursuant to these technical specifications.~~

~~Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.~~

~~j. Deleted.~~

k. ~~Records of Environmental Qualification which are covered under the provisions for paragraph 6.14.~~

~~l. Deleted.~~

~~m. Records of results of analyses required by the Radiological Environmental Monitoring Program.~~

~~n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PLAN.~~

~~o. Records of radioactive shipments~~

~~6.10.3~~ Quality Assurance Records shall be retained as specified by the ***DQAP***~~QATR~~.

6.11 ***DELETED***~~RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

6.12 ***DELETED***~~(Deleted)~~

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by Section 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation at 30 cm (11.8 in.) is greater than deep dose equivalent of 100 mRem/hr but less than 1,000 mRem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).

NOTE: Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are following plant radiation protection procedures for entry into high radiation areas.

An individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive exposure control over the activities within the area and who will perform periodic radiation surveillance at the frequency in the RWP. The surveillance frequency will be established by the management position responsible for radiological controls.

6.13.2 Specification 6.13.1 shall also apply to each high radiation area in which the intensity of radiation is greater than deep dose equivalent of 1,000 mRem/hr at 30 cm (11.8 in.) but less than 500 rads in 1 hour at 1 meter (3.28 ft.) from sources of radioactivity. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of operations and/or radiation protection supervision on duty.

6.14 ***DELETED***~~ENVIRONMENTAL QUALIFICATION~~

~~A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position of Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License DPR-16 dated October 24, 1980.~~

~~B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.~~

6.15 ***DELETED***~~INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT~~

~~The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:~~

~~1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and~~

~~2. System leak test requirements, to the extent permitted by system design and radiological conditions, for each system at a frequency of once every 24 months. The systems subject to this testing are (1) Core Spray, (2) Containment Spray, (3) Reactor Water Cleanup, (4) Isolation Condenser, and (5) Shutdown Cooling.~~

6.16 ***DELETED***~~IODINE MONITORING~~

~~The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas* under accident conditions. This program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for monitoring, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.~~

~~*Areas requiring personnel access for establishing hot shutdown condition.~~

6.17 ***DELETED***~~Deleted~~

6.18 ***DELETED***~~PROCESS CONTROL PLAN~~

~~a. Licensee initiated changes to the PCP:~~

~~1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:~~

~~a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;~~

~~b. a determination that the changes did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and~~

~~c. documentation that the changes have been reviewed and approved pursuant to Section 6.8.2.~~

~~2. Shall become effective upon review and approval by licensee management.~~

6.19 OFFSITE DOSE CALCULATION MANUAL

a. ~~The ODCM shall be approved by the Commission prior to implementation.~~

~~b.~~ Licensee initiated changes to the ODCM shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

1. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

2. a determination that the changes did not reduce the accuracy or reliability of dose calculations or setpoint determination; and,

3. documentation that the changes have been reviewed and approved pursuant to Section 6.8.2.

~~c~~***b***. Change(s) shall become effective upon review and approval by licensee management.

~~6.20 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS~~

DELETED~~.~~

6.21 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license or

   2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.21.b.1 or 6.21.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

~~6.22 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM~~

~~A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room HVAC System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of a 30-day integrated dose of 5 rem TEDE. The program shall include the following elements:~~

~~a. The definition of the CRE and the CRE boundary.~~

~~b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.~~

~~c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.~~

~~The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~

1. ~~The Oyster Creek CRE boundary operability is not dependent on a measured unfiltered air inleakage value (Reference Oyster Creek letter to NRC dated November 17, 2005, Letter No. 2130-05-20218). No inleakage testing for determining the unfiltered air inleakage past the CRE boundary into the CRE is required at the Oyster Creek site.~~

d. ~~Measurement, at designated locations, of the CRE pressure relative to areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem (train) of the Control Room Ventilation System operating at the design flow rate, at a Frequency of 24 months. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

e. ~~The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.~~

f. ~~The provisions of Section 1.24 are applicable to the frequencies for assessing CRE habitability measuring CRE pressure and assessing the CRE boundary as required by paragraphs d and c, respectively.~~

~~6.23 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

a. ~~RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

   i) ~~Limiting Conditions for Operation Section 3.3, "Reactor Coolant"~~
   ii) ~~Surveillance Requirements Section 4.3, "Reactor Coolant"~~

b. ~~The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:~~

   i) ~~SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors"~~

c. ~~The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~

~~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.~~

~~6.25 SNUBBER INSPECTION PROGRAM~~

~~This program conforms to the examination, testing, and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:~~

~~a. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.~~

~~b. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(a), 50.55a(a)(1), 50.55a(a)(1)(i), and 50.55a(a)(1)(iv), subject to its limitations and modifications, and subject to Commission approval.~~

~~c. The program shall, as allowed by 10 CFR 50.55a(b)(3)(v)(B), meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," in lieu of Section XI of the ASME B&PV Code ISI requirements for snubbers, or meet authorized alternatives pursuant to 10 CFR 50.55a(z).~~

~~d. The 120-month program updates shall be made in accordance with 10 CFR 50.55a (including 10 CFR 50.55a(g)(4)(ii)) subject to the conditions listed therein.~~

~~e. Records of the service life of all snubbers, including the date which the service life commences, and associated installation and maintenance records shall be maintained for the duration of the Facility Operating License.~~

Michael P. Gallagher
Exelon Nuclear
Vice President
License Renewal and Decommissioning

200 Exelon Way
Kennett Square, PA 19348

610 765 5958 Office
610 765 5658 Fax
www.exeloncorp.com
michaelp.gallagher@exeloncorp.com

10 CFR 50.90

RA-18-036

March 29, 2018

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
NRC Docket Nos. 50-219 and 72-15

Subject: Response to Request for Additional Information (RAI) and Supplement Regarding License Amendment Request - Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition

Reference:

1) Letter from Michael P. Gallagher, (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – *"License Amendment Request – Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition,"* dated November 16, 2017 (ADAMS Accession No. ML17320A411)

2) U.S. Nuclear Regulatory Commission Electronic Mail Request to David Helker, et al., (Exelon Generation Company, LLC) – DRAFT RAI - Oyster Creek Defueled TS LAR (EPID: L-2017-LLA-0395), dated February 28, 2018

3) U.S. Nuclear Regulatory Commission Electronic Mail Request to David Helker (Exelon Generation Company, LLC) – "RAI for Oyster Creek Permanently Defueled TS LAR and EAL Scheme LAR (EPID: L-2017-LLA-0395)," dated March 9, 2018 (ML18068A658)

4) U.S. Nuclear Regulatory Commission Electronic Mail Request to David Helker, et al., (Exelon Generation Company, LLC) – DRAFT RAI - Oyster Creek Defueled TS LAR (EPID: L-2017-LLA-0395), dated March 19, 2018

By letter dated November 16, 2017 (Reference 1), Exelon Generation Company, LLC (Exelon) submitted changes to Renewed Facility Operating License (RFOL) No. DPR 16 and Technical Specifications (TSs) for the Oyster Creek Nuclear Generating Station (OCNGS). Exelon requested an amendment to revise the OCNGS RFOL and the associated TS to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

Subsequently, in electronic mail requests dated February 28, 2018 (Reference 2), and March 19, 2018 (Reference 4), the U.S. Nuclear Regulatory Commission (NRC) issued draft Requests for Additional Information (RAI) indicating that it had reviewed the information submitted in the Reference 1 letter and that additional clarifying information was needed to support its continued technical review. The draft RAI questions in Reference 2 were further discussed during a teleconference between Exelon and NRC representatives held on March 9, 2018. As a result of the discussion, it was determined that no modifications to the draft RAI questions were needed and the NRC subsequently issued its formal RAI on March 9, 2018 (Reference 3), and requested a response within 30 days of the date of the electronic mail request. The RAI issued in Reference 4 was self-explanatory and did not require further discussion. The response to the last RAI question will be included with the other RAI question responses.

Attachment 1 provides Exelon's responses to the NRC's RAI questions contained in the Reference 3 and Reference 4 electronic mail requests. Attachment 2 includes the revised Permanently Defueled Technical Specifications (PDTS) page mark-ups and Attachment 3 contains the clean pages of the PDTS. The PDTS pages being submitted in this letter supersede in entirety those affected pages submitted in Reference 1.

Exelon has reviewed the information supporting a finding of No Significant Hazards Consideration and the Environmental Consideration provided to the NRC in Reference 1. The additional information provided in this submittal does not affect the previously stated bases in Reference 1 for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this submittal.

If you have any questions concerning this submittal, please contact Paul Bonnett at (610) 765-5264.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of March 2018.

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President, License Renewal & Decommissioning
Exelon Generation Company, LLC

Attachment 1: Response to NRC's Request for Additional Information and Supplement

Attachment 2: Revised Permanently Defueled Technical Specifications (PDTS) Page Mark-ups

Attachment 3: Clean Copy - Permanently Defueled Technical Specifications (PDTS)

cc: w/Attachments

Regional Administrator - NRC Region I
NRC Senior Resident Inspector - Oyster Creek Nuclear Generating Station
NRC Project Manager, NRR - Oyster Creek Nuclear Generating Station
Director, Bureau of Nuclear Engineering - New Jersey Department of Environmental Protection
Mayor of Lacey Township, Forked River, NJ

Attachment 1

Response to NRC's Request for Additional Information and Supplement

SUMMARY

By letter dated November 16, 2017 (Reference 1), Exelon Generation Company, LLC (Exelon) submitted changes to Renewed Facility Operating License (RFOL) No. DPR 16 and Technical Specifications (TSs) for the Oyster Creek Nuclear Generating Station (OCNGS). Exelon requested an amendment to revise the OCNGS RFOL and the associated TS to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

Subsequently, in electronic mail requests dated February 28, 2018 (Reference 2), and March 19, 2018 (Reference 4), the U.S. Nuclear Regulatory Commission (NRC) issued draft Requests for Additional Information (RAI) indicating that it had reviewed the information submitted in the Reference 1 letter and that additional clarifying information was needed to support its continued technical review. The draft RAI questions in Reference 2 were further discussed during a teleconference between Exelon and NRC representatives held on March 9, 2018. As a result of the discussion, it was determined that no modifications to the draft RAI questions were needed and the NRC subsequently issued its formal RAI on March 9, 2018 (Reference 3), and requested a response within 30 days of the date of the electronic mail request. The RAI issued in Reference 4 was self-explanatory and did not require further discussion. The response for the last RAI question will be included with the other RAI question responses.

Accordingly, this attachment restates the NRC's RAI questions contained in the Reference 3 and Reference 4 electronic mail requests followed by Exelon's response. RAI questions 1 through 9 were communicated in Reference 3 and RAI question 10 was communicated in Reference 4. Attachment 2 includes the revised Permanently Defueled Technical Specifications (PDTS) page mark-ups and Attachment 3 contains the clean pages of the PDTS. The PDTS pages being submitted in this letter supersede in entirety those affected pages submitted in Reference 1.

RESPONSE TO RAI QUESTIONS

RAI-(RFOL)-01

You propose to delete RFOL 1.C. In Reference 9 (ML15117A551) of your submittal, you refer to the Vermont Yankee Defueled TS amendment that the NRC approved. In the Vermont Yankee approval, the NRC staff approved the following License Condition, which is similar to your RFOL 1.C that you request to delete.

"Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21 (a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21 (c), such that there is reasonable assurance that the activities authorized by this license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations."

Please provide a technical justification for deleting RFOL 1.C.

Exelon's Response to RAI-(RFOL)-01:

Exelon withdraws its request to delete Licensing Finding 1.C. Exelon reassessed its request to delete this License Finding and concluded that this finding is part of the historical documented

basis for the NRC Staff's conclusions that the license meets other conditions referenced in the license and should remain unaltered in the license until license termination.

RAI-(RFOL)-02

In RFOL 2.C.(2), explain why you wish to delete "incorporated in the license" and replace it with "replaced with the Permanently Defueled Technical Specifications (PDTS)." Once you decommission, you will still have a Part 50 license for OCNGS that you must adhere to.

Exelon's Response to RAI-(RFOL)-02:

Exelon proposed the following change for RFOL 2.C.(2) in Reference 1:

> The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ***[XXX]***, are hereby ~~incorporated in the license~~***replaced with the Permanently Defueled Technical Specifications (PDTS)***. Exelon Generation Company shall ~~operate~~ ***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specifications.

Exelon will revise the proposed change to read:

> The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ***[XXX]***, are hereby incorporated in the license. Exelon Generation Company shall ~~operate~~ ***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specifications ***(PDTS)***.

RAI-(RFOL)-03

On December 22, 2017, the NRC staff issued Amendment No. 292 (ML17289A222) regarding Cyber Security. Your current request would supersede Amendment No. 292. Do you wish to supplement your November 16, 2017, submittal to delete your requested change for RFOL 2.C.(4)?

Exelon's Response to RAI-(RFOL)-03:

Exelon withdraws its request to change for RFOL 2.C.(4). With the NRC issuance of Amendment No. 292 (ML17289A222), the request is no longer valid. A revised page of the RFOL will be supplemented in Attachment 2.

RAI-(RFOL)-04

In RFOL 2.C.(10), you state that Exelon analyzed the FHA for dose results for the CR. Where is this analysis? Is it part of the submittal dated November 16, 2017? Can you provide more technical details of this analysis?

Exelon's Response to RAI-(RFOL)-04:

The following provides additional technical information related to the Fuel Handling Accident (FHA) for post-cessation of power operations (decommissioning FHA) provided in Attachment 1 of Reference 1. Specifically, this response summarizes the changes from the FHA analysis for normal power operation compared to the decommissioning FHA analysis. This information does not supersede the decommissioning FHA information in Attachment 1 of Reference 1. Some of the information is repeated from Attachment 1 of Reference 1 for completeness in summarizing the changes here.

The OCNGS decommissioning FHA analysis (Reference 6) removed credit for safety systems which would reduce offsite and Control Room (CR) doses. The safety systems not credited are (1) the Standby Gas Treatment System (SGTS), (2) secondary containment, and (3) control room ventilation and filtration. With the elimination of these safety systems, a FHA in the Spent Fuel Pool (SFP) would not result in a significant radiological release to the general public or environment. Activity that comes out of the SFP would be released into the refuel floor airspace, then released directly to the environment over a 2-hour time period as per Attachment B in Regulatory Guide 1.183 (RG 1.183) (Reference 7). The released activity then migrates to the CR intake and enters the CR unfiltered. In order to evaluate the consequences of downgrading the safety systems from safety-related to non-safety-related required the FHA analysis to consider whether a malfunction of the systems could affect the CR and offsite doses.

The analysis (Reference 6) determined that a malfunction of the CR ventilation system with no filtration has a substantial impact on the CR dose if the CR ventilation system isolated and contained the activity in the CR. A sensitivity study was performed that determined the worst-case scenario. If the CR ventilation was in purge mode at the time of the FHA (maximizing the intake flow) and the CR ventilation system subsequently fails 5 minutes into the accident isolating the control room for the remaining duration of the event, then the dose to the CR operators is maximized. After CR isolation, 10 cfm airflow into and out of the CR is assumed to account for normal operator ingress and egress.

A new source term for the decommissioning FHA analysis was calculated using ORIGEN2.1 using the highest core average burnup and a bounding core average enrichment applicable to the current operating Cycle 26 and future Cycle 27. Per Reference 5, OCNGS will shutdown at the end of Cycle 26. To assess the dose, a 60-day decay time was assumed and no credit was taken for dose reduction due to safety systems and CR ventilation system isolation since they are being downgraded. ORIGEN2.1 was used to calculate the decay of the fuel for the 60 days. In the FHA analysis for normal power operations, ORIGEN2.1 was used to calculate the source term and RADTRAD was used to decay the fuel 1-day. ORIGEN2.1 is more robust than RADTRAD for calculating the decay greater than a few days as ORIGEN2.1 accounts for the progeny of all parent nuclides while RADTRAD usually uses a limited set of approximately 60 nuclides that are significant to dose.

The activity released from the dropped and struck bundles is provided in Table 1.

Table 1: FHA Noble Gas and Iodine Isotopic Source Terms

Isotope	FHA for Normal Power Operations Specific Activity* (Ci/MWth)	Decommissioning FHA 60 Day Decay Specific Activity* (Ci/MWth)
Kr-85	6.63E+00	7.12E+00
Kr-85m	4.90E+01	0.00E+00
Kr-87	9.35E+01	0.00E+00
Kr-88	1.31E+02	0.00E+00
I-131	1.54E+00	9.10E-03
I-132	1.38E+00	4.07E-06
I-133	1.95E+00	2.87E-21
I-134	2.15E+00	0.00E+00
I-135	1.83E+00	0.00E+00
Xe-133	3.90E+02	1.69E-01
Xe-135	1.65E+02	0.00E+00

* FHA for normal power operations specific activity shown in Table is un-decayed. RADTRAD decay functionality was used to decay 1 day. Decommissioning FHA 60-day decay was performed using ORIGEN2.1 instead of RADTRAD.

The decommissioning FHA can only occur in the SFP. However, for conservatism, the fuel damage value of 2.01 bundles associated with dropping a fuel bundle over the vessel with a fall height of approximately 30 feet (as compared to the SFP drop height of approximately 4 feet) was retained as it maximizes the fuel damage and therefore, the released activity, that could occur during a FHA. A Decontamination Factor (DF) of 200 (associated with 23 feet of water coverage) was used; however, the bundle that lays on the SFP racks has slightly less than 23 feet (approximately 21.5 feet) of water coverage. The use of DF of 200 is acceptable due to the conservatism in the fuel damage (factor of 2 higher) – see response to RAI-(TS 3.1)-09 for more detail.

A ground release atmospheric dispersion factor (X/Q) near the Main Steam Isolation Valves (MSIVs) in the turbine building was used in the decommissioning FHA as compared to a main stack release used in the FHA for normal power operations. This release point was selected because it does not credit any safety-systems, it conservatively bounds the atmospheric dispersion from the fuel building (i.e., reactor building) to the CR intake, and was previously evaluated to address the ground-level MSIV leakage during the Alternate Source Term (AST) Loss of Coolant Accident (LOCA).

RADTRAD version 3.03 (RADTRAD) was used to calculate the offsite and CR dose for the decommissioning FHA and the FHA for normal power operations. Table 2 provides a summary of the changes between the FHA for normal power operations and the decommissioning FHA.

Table 2: Summary of RADTRAD Inputs

Parameter	FHA for Normal Power Operations Value	Decommissioning FHA Value
Core Activities	Refer to Table 1	Refer to Table 1
Environment Release Point	Offsite: Main stack-level release CR: Main stack-level release	Offsite Ground-level from the reactor building CR: MSIVs in the turbine building
Standby Gas Treatment System Filtration and Flowrate	Credited.	Not credited.
Control Room Parameters		
CR Atmospheric Dispersion Factors	0 – 5 min. ground release (2.59E-03 sec/m^3) 5 min. – 8 hours (1.80E-04 sec/m^3) 8 – 24 hours (9.67E-05 sec/m^3) 24 – 96 hours (2.50E-05 sec/m^3) 96 – 720 hours (3.60E-06 sec/m^3)	0 – 8 hours (2.71E-03 sec/m^3) 8 – 24 hours (8.76E-04 sec/m^3) 24 – 96 hours (8.63E-04 sec/m^3) 96 – 720 hours (8.45E-04 sec/m^3)
Unfiltered Intake Flow	0 to 720 hours (1.40E+04 cfm)	0 to 8.3333E-02 hours (1.40E+04 cfm) 8.3333E-02 to 720 hours (10 cfm)

RAI-(TS 3.0)-05

Why does the proposed LCO 3.0.1 refer to TS 3.0.2 and not LCO 3.0.2?

Exelon's Response to RAI-(TS 3.0)-05:

The proposed LCO 3.0.1 referring to TS 3.0.2 is a typographical error and has been corrected to LCO 3.0.1 and LCO 3.0.2. Attachment 2 of this submittal provides the corrected page for Section 3/4, "Limiting Conditions for Operation and Surveillance Requirement Applicability."

RAI-(TS 3.0)-06

Why is there a colon between LCO and 3.1 on page 44 of 91?

Exelon's Response to RAI-(TS 3.0)-06:

The colon between LCO and 3.1 on page 44 of 91 (and also between LCO and 3.2 on page 55 of 91) represent the formatting as provided in the actual LCOs provide on pages 3/4/1-1 and 3/4/2-1 respectively, where "LCO:" is intended to highlight the LCO.

RAI-(SR 4.2)-07

Why is the frequency once per 7 days when liquid is being added? Why not once per 7 days while radioactive liquid is in the tank?

Exelon's Response to RAI-(SR 4.2)-07:

The surveillance frequency in proposed SR 4.2 is based on the frequency of SR 4.6.C in the current operating TS. The description and frequency of SR 4.6.C states:

> Liquids contained in the following tanks shall be sampled and analyzed for radioactivity.
> Once per 7 days when radioactive liquid is being added to the tank.

LCO 3.2 requires action in the event the quantity of radioactive material in any applicable storage tank exceeds 10.0 curies. The tank is sampled once per 7 days when radioactive liquid is being added to the tank to ensure the total quantity of radioactive material does not exceed 10 curies. If it does then the Required Action of the LCO to begin and continue treatment until the quantity to 10 curies or less. If the existing material in the tank is less than 10 curies, the only means that would cause the radioactivity to increase would be by adding liquid to the tank. Therefore, taking a sample every 7-days when additions are being made to the tank verifies that the radioactivity will remain within limits. A surveillance once per 7 days would not be meaningful if there are no additions being made to the tank.

RAI-(TS 6.0)-08

You are proposing to change TS 6.8.1.a, TS 6.8.4.a.9, TS 6.9.1.d, TS 6.9.1.e; however, Amendment No. 290 has been approved with a 60-day implementation period not to exceed March 29, 2020. These are the same TSs being revised in the Defueled TS LAR. How is Exelon going to ensure that Amendment No. 290 is implemented before the Defueled TS amendment? What controls does Exelon have in place?

Exelon's Response to RAI-(TS 6.0)-08:

Exelon is tracking the approval of each amendment request and will implement each amendment sequentially as it was issued by the NRC. Amendment 290 revised and removed certain requirements from the Section 6, "Administrative Controls," that are not applicable to the facility in a permanently defueled condition. Specifically, the amendment revised TS Section 6.1, "Responsibility"; TS Section 6.2, "Organization"; TS Section 6.3, "Facility Staff Qualifications"; TS Section 6.6, "Reportable Event Action"; TS Section 6.7, "Safety Limit Violation"; TS Section 6.8, "Procedures and Programs"; and TS Section 6.9, "Reporting Requirements" to reflect the staffing and training requirements for operating staff when the facility is permanently defueled.

Amendment No. 290 will be implemented immediately after the certification of permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii), has been docketed with the NRC.

The proposed License Amendment Request (LAR) is to revise the OCNGS Renewed Facility Operating License (RFOL) and associated TS to Permanently Defueled Technical Specification (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. This PDTS LAR will be implemented after a 60-day period from the permanent unit shutdown to allow sufficient decay of the fuel in order to ensure that radiological doses at the EAB, LPZ, and in the Control Room from a postulated FHA are below the limits of 10 CFR 50.67 and Regulatory Guide 1.183.

Amendment (AMD) No. 290 will be implemented at shutdown. This proposed PDTS LAR will not be implemented until after 60 days subsequent to shutdown, ensuring that it is implemented after AMD No 290. This implementation sequence is being tracked in the action tracking system to ensure the timely and effective implementation of the amendments.

RAI-(TS 3.1)-09

On April 26, 2007, the NRC issued Amendment No. 262 (ADAMS No. ML071080019) to Facility Operating License No. DPR-16 for the OCNGS, approving the implementation of the alternative source term in accordance with 10 CFR 50.67 following the guidance provided in applicable sections of Regulatory Guide (RG) 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*.

Chapter 15 of the OCNGS Updated Final Safety Analysis Report (UFSAR) describes the design basis accidents (DBA) and transient scenarios applicable to OCNGS during power operations. With the termination of reactor operations at OCNGS and the permanent removal of fuel from the reactor pressure vessel (RPV) the majority of the DBA scenarios postulated in the UFSAR are no longer possible. During decommissioning, the irradiated fuel will be stored in the Spent Fuel Pool (SFP) and the Independent Spent Fuel Storage Installation until it is shipped off site. The analyzed accidents that remain applicable to OCNGS in the permanently shut down and defueled condition is a Fuel Handling Accident (FHA) in the SFP (a dropped fuel assembly onto the top of the core will no longer be applicable), the Postulated Radioactive Tank Failure, and Release of Radioactive Liquid Waste while radioactive liquids are still present.

The licensee proposed to change TS Section 3.1, *Protective Instrumentation* to be TS Section 3/4.1, *Spent Fuel Storage* with a new proposed TS LCO: TS 3.1, *Spent Fuel Pool Water Level*. The purpose of the change is to ensure safe storage and management of the spent fuel. LCO 3.1, *Spent Fuel Pool Water Level*, specifies requirements to ensure that the minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a FHA in the SFP. LCO 3.1 states:

> "Whenever irradiated fuel is stored in the spent fuel storage pool, water level shall be maintained at a level ≥ 117 feet 8 inches (elevation above sea level) with the exception of planned cask movements."

To support this new proposed change, the licensee evaluated the required minimum water level in the SFP (calculation C-1302-226-E310-460) for the dropping an irradiated fuel assembly onto irradiated fuel bundles stored in the SFP.

The licensee stated:

> "There is slightly over 23 feet of water above the top of active fuel for bundles within the storage racks; however, this does not ensure the dropped bundle will have 23 feet of water coverage above it."

Regulatory Issue Summary (RIS) 2006-04, Summary of Issue 8, *Elemental Iodine Decontamination Factor (DF),* explains:

> "Appendix B to RG 1.183, provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above [emphasis added] the damaged fuel is 23 feet or greater, Regulatory Position 2 states that the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500."

The licensee credits an overall effective decontamination factor of 200 even though 23 feet of water above the dropped bundle is not ensured. The licensee justifies the decontamination factor:

> "because less fuel damage would occur due to the shorter drop compensating for having slightly less than 23 feet of water coverage above the dropped bundle."

It is unclear how a decontamination factor of 200 is justified due to the reduced drop height in the SFP. The licensee's fuel assembly drop analysis uses a simplified conservation of energy approach to estimate the maximum kinetic energy generated from a dropped fuel assembly. The analysis does not estimate a decontamination factor due to the assembly drop height.

As discussed in RG 1.183, Regulatory Position 2, *Water Depth,* if the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.

Please provide the technical basis for applying a decontamination factor associated with the water above the spent fuel being 23 feet or greater, when the water depth in the SFP is not 23 feet or greater. In addition, explain the methodology used to calculate the decontamination factor. Please state the estimated water level [in linear feet] above the damaged fuel assembly as applied in the radiological consequence analysis.

Exelon's Response to RAI-(TS 3.1)-09:

Once the reactor is decommissioned, the FHA can only occur in the SFP. A pool DF of 200 was used, however, the bundle that lays on the SFP racks has approximately 21.5 linear feet of water coverage and the spent fuel assemblies within the storage racks have approximately 23 linear feet of water coverage above the top of active fuel. The reason why the use of a DF of 200 is acceptable, even though the dropped bundle does not have 23 feet of water coverage, is because the FHA used a conservative bundle damage fraction associated with a bundle drop over the RPV. A fuel damage value of 2.01 damaged bundles associated with dropping a fuel bundle over the RPV with a fall height of approximately 30 feet is conservative compared to 50% less fuel damage for the SFP drop with a height of approximately 4 feet. The fuel damage with the RPV drop and the use of a DF of 200 conservatively releases more activity during the FHA than the 50% less fuel damage for the SFP drop with a lower DF accounting for the less water coverage above the dropped bundle. Therefore, the RPV drop with the use of a DF of 200 was conservatively used in the FHA analysis.

RAI-(TS 1.0)-10

In Attachment 1, Exelon proposes to delete definitions as described on pages 23 through 32. In Attachment 2, Exelon marks up pages 1.0-1 and 1.0-9. Where are the mark-ups of pages 1.0-2 through 1.0-8 to match Attachment 1 descriptions?

Exelon's Response to RAI-(TS 3.1)-10:

Pages 1.0-2 through 1.0-8 of Section 1, Definitions, were not submitted in Attachment 2 of Reference 1 because all the definitions on these pages were proposed for deletion. As noted in Reference 1, for TSs that are deleted in their entirety, Exelon chose not to include the deleted pages in Attachment 2. However, these pages are provided in Attachment 2 of this submittal.

REFERENCES:

1. Letter from Michael P. Gallagher, (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – *"License Amendment Request – Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition,"* dated November 16, 2017 (ADAMS Accession No. ML17320A411)

2. U.S. Nuclear Regulatory Commission Electronic Mail Request to David Helker, et al., (Exelon Generation Company, LLC) – *"DRAFT RAIs - Oyster Creek Defueled TS LAR (EPID: L-2017-LLA-0395),"* dated February 28, 2018

3. U.S. Nuclear Regulatory Commission Electronic Mail Request to David Helker (Exelon Generation Company, LLC) – *"RAI for Oyster Creek Permanently Defueled TS LAR and EAL Scheme LAR (EPID: L-2017-LLA-0395),"* dated March 9, 2018 (ADAMS Accession No. ML18068A658)

4. U.S. Nuclear Regulatory Commission Electronic Mail Request to David Helker, et al., (Exelon Generation Company, LLC) – *"DRAFT RAIs - Oyster Creek Defueled TS LAR (EPID: L-2017-LLA-0395),"* dated March 19, 2018

5. Letter from Michael P. Gallagher, Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission – *“Certification of Permanent Cessation of Power Operations for Oyster Creek Nuclear Power Station,”* dated February 14, 2018 (ADAMS Accession No. ML18045A084)

6. C-1302-226-E310-460, "EAB, LPZ, and CR Doses Due to Fuel Handling Accident (FHA) – Post Cessation of Power Operations," August 9, 2017

7. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000, (ADAMS Accession No. ML003716792)

Attachment 2

Revised Permanently Defueled Technical Specifications (PDTS) Page Mark-ups

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-16, has now found that:

   A. The application for a Renewed Facility Operating License No. DPR-16 filed by the applicant 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 and all required notifications to other agencies or bodies have been duly made;

   B. ***DELETED*** ~~Construction of the Oyster Creek Nuclear Generating Station (Oyster Creek or the facility) has been completed in conformity with Provisional Construction Permit No. CPPR-15; the application, as amended; the provisions of the Act; and the rules and regulations of the Commission.~~

   C. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the term of this Renewed Facility Operating License No. DPR-16 on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;

   D. The facility will ***be maintained*** ~~operate~~ in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission ~~(except as exempted from compliance in Section 2.D. below)~~;

(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 ***that were used*** for reactor startup, sealed sources ***that were used*** for ***calibration of*** reactor instrumentation and ***are used in*** 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~~ ***that were*** 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) ***DELETED***~~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. ***[###]***, are hereby incorporated in the license. Exelon Generation Company shall ~~operate~~***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specificationsn ***(PDTS)***.

(3) ***DELETED***~~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.~~

(4) Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Oyster Creek Nuclear Generating Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 5." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by Renewed License Amendment No. 280 and modified by License Amendment No. 288 and 292.

(5) ***DELETED***~~Inspections of core spray spargers, piping and associated components will be performed in accordance with BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," as approved by NRC staffs Final Safety Evaluation Report dated December 2, 1999.~~

(6) ***DELETED***~~Long Range Planning Program – Deleted~~

(7) ***DELETED***~~Reactor Vessel Integrated Surveillance Program~~

~~Exelon Generation Company is authorized to revise the Updated Final Safety Analysis Report (UFSAR) to allow implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel Integrated Surveillance Program as the basis for demonstrating compliance with the requirements of Appendix H to Title 10 of the *Code of Federal Regulations* Part 50, "Reactor Vessel Material Surveillance Program Requirements," as set forth in the licensee's application dated December 20, 2002, and as supplemented on May 30, September 10, and November 3, 2003.~~

~~All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program appropriate for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.~~

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

SECTION I
DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of the specifications.

1.1 ~~OPERABLE-OPERABILITY~~ ***ACTIONS***

ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.

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

~~A verification of operability is an administrative check, by examination of appropriate plant records (logs, surveillance test records) to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine operability.~~

1.2 ~~OPERATING~~ ***CERTIFIED FUEL HANDLER***

~~Operating means that a system or component is performing its required function.~~

1.3 ~~POWER OPERATION~~ ***NON-CERTIFIED OPERATOR***

~~Power operation is any operation when the reactor is in the startup mode or run mode except when primary containment integrity is not required.~~

~~1.4 STARTUP MODE~~

~~The reactor is in the startup mode when the reactor mode switch is in the startup mode position. In this mode, the reactor protection system scram trips initiated by condenser low vacuum and main steam line isolation valve closure are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure is bypassed; the IRM trips for rod block and scram are operable; and the SRM trips for rod block are operable.~~

~~1.5 RUN MODE~~

~~The reactor is in the run mode when the reactor mode switch is in the run mode position. In this mode, the reactor protection system is energized with APRM protection and the control rod withdrawal interlocks are in service.~~

~~1.6 SHUTDOWN CONDITION~~

~~The reactor is in the SHUTDOWN CONDITION when there is fuel in the reactor vessel, the reactor is subcritical, all operable control rods are fully inserted, and the mode switch is in the shutdown mode position. In this position, a control rod block is initiated.~~

~~1.7 COLD SHUTDOWN CONDITION~~

~~The reactor is in the COLD SHUTDOWN CONDITION when the reactor is in the SHUTDOWN CONDITION, and (except during REACTOR VESSEL PRESSURE TESTING), the reactor coolant system is maintained at less than 212°F and vented.~~

~~1.8 PLACE IN SHUTDOWN CONDITION~~

~~Proceed with and maintain an uninterrupted normal plant shutdown operation until the SHUTDOWN CONDITION is met.~~

~~1.9 PLACE IN COLD SHUTDOWN CONDITION~~

~~Proceed with and maintain an uninterrupted normal plant shutdown operation until the COLD SHUTDOWN CONDITION is met.~~

~~1.10 PLACE IN ISOLATED CONDITION~~

~~Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.~~

~~1.11 REFUEL MODE~~

~~The reactor is in the REFUEL MODE when the reactor mode switch is in the REFUEL MODE position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.~~

~~1.12 REFUELING OUTAGE~~

~~For the purpose of designating frequency of testing and surveillance, a REFUELING OUTAGE shall mean a regularly scheduled REFUELING OUTAGE. Following the first REFUELING OUTAGE, successive tests or surveillances shall be performed at least once per 24 months.~~

~~1.13 PRIMARY CONTAINMENT INTEGRITY~~

~~PRIMARY CONTAINMENT INTEGRITY means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied:~~

~~A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.~~
~~B. At least one door in the airlock is closed and sealed.~~
~~C. All automatic primary containment isolation valves are OPERABLE or the affected penetration is isolated.~~
~~D. All blind flanges and manways are closed.~~

~~1.14 SECONDARY CONTAINMENT INTEGRITY~~

~~Secondary containment integrity means that the reactor building is closed and the following conditions are met:~~

~~A. At least one door at each access opening is closed.~~
~~(Note: Momentary opening and closing of the trunnion room door does not constitute a loss of secondary containment intergrity. In COLD SHUTDOWN CONDITION or REFUEL MODE, the trunnion room door may remain open provided the trunnion room is isolated from the secondary containment through the reactor building walls, penetrations and either the inboard or outboard valves to the main steam and feedwater piping being secured in the closed position.)~~

~~B. The standby gas treatment system is operable.~~

~~C. All automatic secondary containment isolation valves are operable or are secured in the closed position.~~

~~1.15 (DELETED)~~

~~1.16 RATED FLUX~~

~~Rated flux is the neutron flux that corresponds to a steady state power level of 1930 MW(t). Use of the term 100 percent also refers to the 1930 thermal megawatt power level.~~

~~1.17 REACTOR THERMAL POWER-TO-WATER~~

~~Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.~~

~~1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS~~

~~A. Instrument Channel~~

~~An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.~~

~~B. Trip System~~

~~A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).~~

~~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.~~

~~1.28 FRACTION OF RATED POWER (FRP)~~

~~The FRACTION OF RATED POWER is the ratio of core THERMAL POWER to RATED THERMAL POWER.~~

~~1.29 TOP OF ACTIVE FUEL (TAF) - 353.3 inches above vessel zero.~~

~~1.30 REPORTABLE EVENT~~

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

~~1.31 IDENTIFIED LEAKAGE~~

~~IDENTIFIED LEAKAGE is that leakage which is collected in the primary containment equipment drain tank and eventually transferred to radwaste for processing.~~

~~1.32 UNIDENTIFIED LEAKAGE~~

~~UNIDENTIFIED LEAKAGE is all measured leakage that is other than identified leakage.~~

~~1.33 PROCESS CONTROL PLAN~~

~~The PROCESS CONTROL PLAN shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

~~1.34 AUGMENTED OFFGAS SYSTEM (AOG)~~

~~The AUGMENTED OFFGAS SYSTEM is a system designed and installed to holdup and/or process radioactive gases from the main condenser offgas system for the purpose of reducing the radioactive material content of the gases before release to the environs.~~

~~1.35 MEMBER OF THE PUBLIC~~

~~A MEMBER OF THE PUBLIC is a person who is not occupationally associated with Exelon Generation Company, LLC and who does not normally frequent the Oyster Creek Nuclear Generating Station site. The category does not include contractors, contractor employees, vendors, or persons who enter the site to make deliveries, to service equipment, work on the site, or for other purposes associated with plant functions.~~

~~1.36 OFFSITE DOSE CALCULATION MANUAL (ODCM)~~

~~The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and~~

~~parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4; and (2) descriptions of the information that should be included in the Annual Radioactive Effluent Release Report AND Annual Radiological Environmental Operating Report required by Specifications 6.9.1.d and 6.9.1.e, respectively.~~

~~1.37 PURGE~~

~~PURGE OR PURGING is the controlled process of discharging air or gas from a confinement and replacing it with air or gas.~~

~~1.38 SITE BOUNDARY~~

~~The SITE BOUNDARY is the perimeter line around the OCNGS beyond which the land is neither owned, leased nor otherwise subject to control by Exelon Generation Company, LLC (ref. ODCM). The area outside the SITE BOUNDARY is termed OFFSITE or UNRESTRICTED AREA.~~

~~1.39 REACTOR VESSEL PRESSURE TESTING~~

~~System pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely water solid, core not critical and section 3.2.A satisfied.~~

~~1.40 SUBSTANTIVE CHANGES~~

~~SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Example of non-substantive changes are: (1) correcting spelling, (2) adding (but not deleting) sign-off spaces, (3) blocking in notes, cautions, etc, (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications, and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.~~

~~1.41 DOSE EQUIVALENT I-131~~

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 microcuries per gram which alone would produce the same thyroid dose as the quantity and isotopic mixture of I131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 or Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluences for the Purpose of Evaluating Compliance with 10 CFR Par 40 Appendix I."~~

~~1.42 AVERAGE PLANAR LINEAR HEAT GENERATION RATE~~

~~The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.~~

~~1.43 CORE OPERATING LIMITS REPORT~~

~~The Oyster Creek CORE OPERATING LIMITS REPORT (COLR) is the document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.f. Plant operation within these operating limits is addressed in individual specifications.~~

~~1.44 LOCAL LINEAR HEAT GENERATION RATE~~

~~The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) shall be applicable to a specific planar height and is equal to the AVERAGE PLANAR LINEAR GENERATION RATE (APLHGR) at the specified height multiplied by the local peaking factor at that height.~~

~~1.45 SHUTDOWN MARGIN (SDM)~~

~~SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:~~

~~a. The reactor is xenon free;~~
~~b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and~~
~~c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.~~

~~1.46 IDLE RECIRCULATION LOOP~~

~~A recirculation loop is idle when its discharge valve is in the closed position and its discharge bypass valve and suction valve are in the open position.~~

~~1.47 ISOLATED RECIRCULATION LOOP~~

~~A recirculation loop is fully isolated when the suction valve, discharge valve and discharge bypass valve are in the closed position.~~

~~1.48 OPERATIONAL CONDITION~~

~~The reactor plant operational status as to criticality, reactor mode switch position, reactor coolant temperature, and/or specific system status. These conditions consist of POWER~~

~~OPERATION, STARTUP MODE, SHUTDOWN CONDITION, COLD SHUTDOWN CONDITION, and REFUEL MODE. A change or entry into an operating condition is Signified by movement of the reactor mode switch or a change in reactor coolant Temperature from <212°F to ≥212ºF.~~

~~1.49 RATED THERMAL POWER (RTP)~~

~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.~~

~~1.50 THERMAL POWER~~

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

~~1.51 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.23.~~

1.~~5~~2 CERTIFIED FUEL HANDLER

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.~~5~~3 NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

SECTION 3/*4*

LIMITING CONDITIONS FOR OPERATION ***AND SURVEILLANCE REQUIREMENTS***

3/*4*.0 LIMITING CONDITIONS FOR OPERATION ~~(GENERAL)~~ ***AND SURVEILLANCE REQUIREMENT APPLICABILITY***

Applicability: Applies to all Limiting Conditions for Operation ***and Surveillance Requirements.***

Objective: To preserve the single failure criterion for safety systems.

LCO Applicability:

LCO 3.0.1 ***LCOs shall be met during the specified conditions in the TS, except as provided in ~~TS~~LCO 3.0.2.***

LCO 3.0.2 ***Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.***

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

~~A. In the event Limiting Conditions for Operation (LCOs) and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable. Exceptions to the requirements shall be stated in the individual specifications.~~

~~B. When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of applicable LCOs., provided (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfieid, the unit shall be placed in COLD SHUTDOWN within the following 30 hours or within the time specified in the applicable specification. This specification is not applicable in COLD SHUTDOWN or the REFUEL MODE.~~

~~C. When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:~~

~~1. When the associated LCO requirement permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or~~

~~2. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the applicability, and the establishment of risk management actions, if appropriate; exceptions to this specification are stated in the individual Specifications, or~~

~~3. When an allowance is stated in the individual value, parameter, or other Specification.~~

~~This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.~~

Attachment 3

Clean Copy - Permanently Defueled Technical Specifications (PDTS) pages

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

RENEWED FACILITY OPERATING LICENSE

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   B. DELETED

   C. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the term of this Renewed Facility Operating License No. DPR-16 on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;

   D. The facility will be maintained in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

(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 that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment, 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 that were 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) DELETED

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ***[###]***, are hereby incorporated in the license. Exelon Generation Company shall maintain the facility in accordance with the Permanently Defueled Technical Specificationsn (PDTS).

(3) DELETED

(4) Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Oyster Creek Nuclear Generating Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 5." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by Renewed License Amendment No. 280 and modified by License Amendment No. 288 and 292.

(5) DELETED

(6) DELETED

(7) DELETED

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

SECTION I
DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of the specifications.

1.1 ACTIONS

ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.

1.2 CERTIFIED FUEL HANDLER

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.3 NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

SECTION 3/4

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENT APPLICABILITY

Applicability: Applies to all Limiting Conditions for Operation and Surveillance Requirements.

Objective: To preserve the single failure criterion for safety systems.

LCO Applicability:

LCO 3.0.1 LCOs shall be met during the specified conditions in the TS, except as provided in LCO 3.0.2.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

Surveillance Requirement Applicability

SR 4.0.1 Surveillance requirements shall be met during the specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in 4.0.2. Surveillances do not have to be performed onvariables outside specified limits.

SR 4.0.2 If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

SR 4.0.3 Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillance has been met within its specified frequency, except as provided by 4.0.2.

This provision shall not prevent entry into other specified conditions in the Applicability that are required to comply with LCO requirements or that are part of a shutdown of the unit.

SR 4.0.4 The specified frequency for each SR is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.