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10 CFR 50.90

RS-16-228

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

Clinton Power Station
Facility Operating License No. NPF-62
NRC Docket Nos. 50-461and 72-1046

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

References:

1. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Certification of Permanent Cessation of Power Operations," dated June 20, 2016 (ML16172A137)

2. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition," dated July 28, 2016 (ML16210A300)

3. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition – Supplement 1," dated November 4, 2016 (ML16309A013)

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)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit" Exelon Generation Company, LLC (EGC) requests amendment to the Facility Operating License (FOL) and Appendix A, Technical Specifications (TS), of FOL No. NPF-62 for Clinton Power Station (CPS). The proposed amendment would revise the FOL 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 June 2, 2016, EGC announced that it plans to close CPS, Unit 1 due to deteriorating economics. Under the terms of this announcement, EGC agreed to permanently cease operations at CPS by June 1, 2017. By letter dated June 20, 2016 (Reference 1), EGC provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.4(b)(8) and 10 CFR 50.82(a)(1)(i) of EGC's determination to permanently cease operations at CPS by June 1, 2017.

Once the certifications for permanent cessation of operations and removal of fuel from the reactor vessel are docketed for CPS in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the 10 CFR 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. In support of this condition, the CPS FOL 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).

Reference 2 proposed changes to the staffing and training requirements for the CPS staff contained in Section 5.0, Administrative Controls, of the CPS TS to reflect the revised station organization, including Certified Fuel Handlers, which will be implemented once CPS is permanently defueled. This submittal was supplemented on November 4, 2016 (Reference 3). By letter dated September 6, 2016, the NRC approved the Certified Fuel Handler training program (Reference 4).

Attachment 1 to this letter provides a detailed description and evaluation of the proposed changes to the FOL and TS. Attachment 2 contains a markup of the current FOL and TS pages (TS sections that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 2). Attachment 3 contains a markup of the Bases pages. The Bases changes are provided for information only. Since the changes to the TS are considered a major rewrite, revision bars are not used.

EGC has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92, "Issuance of amendment."

The proposed changes have been reviewed and approved by the CPS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests review and approval of this proposed amendment by November 17, 2017, to support the current schedule for the CPS transition to a permanently defueled facility. EGC requests that the approved amendment become effective upon NRC approval of this proposed amendment, or 60 days following the docketing of the certifications required by 10 CFR 50.82(a)(1) that CPS has been permanently shutdown and defueled, whichever is later. The amendment shall be implemented within 90 days from the effective date of the amendment. This timeframe will allow for CPS to complete defueling operations, provides for sufficient decay of the reactor fuel to occur, and implement the proposed TS amendment.

There are no regulatory commitments contained within this submittal.

In accordance with 10 CFR 50.91 "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois 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 Mr. Paul Bonnett at (610) 765-5264.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of November 2016.

Respectfully,

Michael P. Gallagher

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

Attachments:
1. Evaluation of Proposed Changes
2. Proposed Technical Specifications (Marked-Up Pages)
3. Proposed Technical Specifications Bases (Marked-Up Pages)

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Clinton Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

Attachment 1

License Amendment Request

Clinton Power Station

Docket Nos. 50-461 and 72-1046

EVALUATION OF PROPOSED CHANGES

Subject: Proposed Changes to 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 Facility Operating License (FOL) and Appendix A, Technical Specifications (TS), of FOL No. NPF-62 for Clinton Power Station (CPS). The proposed amendment would revise the FOL 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 June 2, 2016, Exelon announced that it plans to close CPS, Unit 1 due to deteriorating economics. Under the terms of this announcement, Exelon agreed to permanently cease operations at CPS by June 1, 2017. By letter dated June 20, 2016 (Reference 1), Exelon provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.4(b)(8) and 10 CFR 50.82(a)(1)(i) of Exelon's determination to permanently cease operations at CPS by June 1, 2017.

Once the certifications for permanent cessation of operations and removal of fuel from the reactor vessel are docketed for CPS in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the 10 CFR 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. In support of this condition, the CPS FOL 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) “Decommissioning.”

On July 28, 2016, Exelon submitted proposed changes to the staffing and training requirements for the CPS staff contained in Section 5.0, Administrative Controls, of the CPS TS (Reference 2) to reflect the revised station organization, including Certified Fuel Handlers, which will be implemented once CPS is permanently defueled. This submittal was supplemented on November 4, 2016 (Reference 3). By letter dated September 6, 2016, the NRC approved the Certified Fuel Handler training program (Reference 4).

The proposed PDTS changes revise and remove certain requirements contained within the FOL and TS, and remove the requirements that would no longer be applicable after all of the fuel has been permanently removed from the CPS reactor in accordance with 10 CFR 50.82(a)(1)(ii). The proposed changes to the FOL and TS 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 CPS 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.

In the development of the proposed PDTS changes, Exelon reviewed the PDTS requirements from other plants that have permanently shutdown, primarily Vermont Yankee (Reference 7), Kewaunee (Reference 8), San Onofre Nuclear Generating Station (Reference 9), Zion Nuclear

Power Station (Reference 10), and Crystal River Nuclear Plant, Unit 3 (Reference 11). Exelon also evaluated the applicable guidance in NUREG-1434, "Standard Technical Specifications - General Electric Plants (BWR/6)" (Reference 6).

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

Pending Licensing Actions under NRC Review

There are several other pending licensing actions currently under NRC review that complement and support the changes proposed by this License Amendment Request (LAR). For clarity, the marked-up FOL and TS pages included in Attachment 2 reflect the changes proposed by these pending actions. In Reference 2, proposed changes to staffing and training requirements contained in CPS TS Section 5.0, Administrative Controls were submitted and supplemented (Reference 3) to reflect the revised station organization, which includes Certified Fuel Handlers (CFH). Reference 14 proposed changes to remove the "Inservice Testing Program" in TS 5.5.6. The NRC has approved the CFH training and retraining program (Reference 4), which will be implemented once all of the fuel has been permanently removed from the CPS reactor and the certification of permanent removal of fuel from the reactor vessel, has been docketed in accordance with 10 CFR 50.82(a)(2).

There are currently two other pending license amendment requests involving proposed changes to TS currently docketed for CPS. One proposed amendment is a request to revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program"; and the other proposed amendment is a request to revise TS Limiting Condition for Operation (LCO) 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." On August 11, 2016, Exelon requested the NRC temporarily suspend review of the two LARs until an evaluation of the need for the LARs can be completed (Reference 5). Exelon expects that this evaluation will be completed prior to the end of 2016.

2.0 DETAILED DESCRIPTION AND BASIS FOR THE CHANGES

The proposed amendment would modify the CPS license and revise CPS TS into PDTS to comport with a permanently defueled condition.

General Analysis Applicable to Proposed Change

Chapter 15 of the CPS Updated Safety Analysis Report (USAR) describes the design basis accidents (DBAs) and transient scenarios applicable to CPS during power and refueling operations and the accidents with the greatest potential for radiation exposure. 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 USAR involve failures or malfunctions of the reactor, RCS, steam system, turbine generator, and emergency core cooling systems (ECCS) which could affect the reactor core.

However, once the certifications for permanent cessation of operations and removal of fuel from the reactor vessel are docketed for CPS 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 USAR will no longer be applicable. The irradiated fuel will be stored in the spent fuel pool (SFP) in the Fuel Building 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 Irradiated Fuel Management Plan.

The postulated DBAs that will remain applicable to CPS in its permanently shutdown and defueled condition are the fuel handling accident (FHA) in the SFP, the Postulated Radioactive Releases Due to Liquid Radwaste Tank Failures, and the Cask Drop Accident. The Cask Drop Accident is no longer a credible accident since the Fuel Building Crane has been upgraded to meet single-failure proof requirements in accordance with NUREG-0612.

On July 5, 2001, Exelon submitted a License Amendment Request to revise the TS requirements that apply during the movement of irradiated fuel and during Core Alterations. Specifically, the proposed changes relaxed the operability requirements for primary containment systems, secondary containment systems, and the Standby Gas Treatment (SGT) system due to a revised analysis of the FHA for CPS based on the implementation of the alternative source term (AST) (Reference 12). The NRC issued the associated License Amendment on April 3, 2002 (Reference 13). This is the current CPS analysis of record (AOR) for the FHA. Following 24 hours of decay time after reactor shut down, the CPS FHA AOR concluded that with the exception of control room ventilation and AC systems, that the dose consequences are acceptable without relying on any other structures, systems, and components (SSCs) remaining operable for accident mitigation during and following the event.

As such, the FHA AOR did not support the revision of TS Limiting Condition for Operation (LCOs) 3.7.3, “Control Room Ventilation System” and 3.7.4, “Control Room Air Conditioning (AC) System.” Specifically, the TS LCO Applicability was not changed from “movement of irradiated fuel” to “movement of *recently* irradiated fuel.” This was because of a key assumption for the FHA analysis, which stated, “the dual inlet control room ventilation is used including the radiation monitors at the inlets consistent with the requirements of the control room ventilation TS.” A new FHA Evaluation was performed as discussed below to support removal the requirements for TS LCOs 3.7.3 and 3.7.4.

Once the control room ventilation/AC TS LCOs are eliminated, the TS LCOs for other systems that support the control room ventilation/AC systems will no longer be needed and are being proposed for deletion. The support systems include a majority of the TS LCOs in TS Section 3.8 Electric Power System.

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. In a permanently defueled condition, the scope of equipment and parameters that must be included in the CPS TS 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

In the permanently defueled condition, the Chapter 15 accident that remains applicable to determining main control room radiological dose is the FHA. The FHA AOR determined the projected dose due to the drop of a fuel assembly onto other fuel assemblies as a function of time after shut down. It demonstrates that radiological doses at the exclusion area boundary, low population zone, and in the control room from a FHA after 24 hours following shut down are within allowable limits. Primary and Secondary containment operability and the standby gas treatment system operation, or control room air intake or recirculation filtration were not credited. However, one of the key assumption for the FHA AOR stated that “the dual inlet control room ventilation is used including the radiation monitors at the inlets consistent with the requirements of the control room ventilation TS.”

To support this proposed LAR, an analysis (IP-F-0179, Site Boundary and Control Room Dose Following a FHA in Containment – Post Cessation of Power Operations) was performed to assess the dose consequences of a postulated FHA after cessation of power operations. This analysis was performed using the methodology currently described in the CPS USAR, which is based on the Alternative Source Term (AST) defined in Regulatory Guide 1.183.

This analysis determined the consequences resulting from an FHA without credit of safety-related SSCs currently in the USAR analysis, including the control room and AC systems in TS 3.7.3 and 3.7.4. The only mitigation credited in the analysis is due to the decontamination action provided by the water covering the damaged fuel. Thus the requirement of 23 feet of water covering the fuel in the spent fuel pool is retained in TS 3.7.7. The analysis determined the required amount of radioactive decay to ensure the dose consequences are within the limits prescribed in 10 CFR 50.67 as supplemented by Regulatory Guide 1.183.

The accident is assumed to occur during handling of a spent fuel assembly. The spent fuel assembly is assumed to drop and strike assemblies stored in the spent fuel pool. As in the current USAR analysis, the amount of fuel that fails is assumed to correspond to a dropped fuel assembly from 34 feet onto the reactor core during refueling. This amount of fuel damage is significantly higher than a drop in the spent fuel pool, which corresponds to a fall of six feet. The damaged fuel assemblies are assumed to release the gaseous gap activity associated with noble gases and iodine. The released activity instantaneously releases and passes through the 23 feet of water in the spent fuel pool. The water provides a decontamination effect to retain most of the iodine. 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 conservatively bounds the atmospheric dispersion from the fuel building to any control room intake and was previously evaluated to address the ground-level MSIV leakage during the AST Loss of Coolant Accident (LOCA) analysis.

The control room 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 control room ventilation model. To address downgrading of safety-related features of the control room ventilation system, a sensitivity study on flow rate was performed. This study determined that the maximum control room dose consequences occur after the control room is completely isolated (no inflow or outflow) at 31 minutes post-accident. This assumption ensures dose rates that bound any scenario where the control room ventilation system stops and traps activity inside. It is noted that this assumption is very conservative and does not factor in the flow due to ingress and egress, which would reduce the airborne activity trapped inside the control room.

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	4.61	5
Exclusion Area Boundary	2.56E-3	6.3
Low Population Zone	5.85E-4	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 HVAC system actuations. The analysis conservatively assumes a large unfiltered in-leakage into the control room and also accounts for unplanned stoppage of the ventilation flow in and out of the control room.

A detailed summary of the inputs used in this analysis are provided below:

Parameter	**Current AOR**	**Revised Analysis**
Reactor Power (MWth)	3473	3473
No. of Damaged Rods	172	172
Equivalent Full Length Rods per Fuel Bundle	85.6	85.6
No. of Bundles in Core	624	624
Radial Peaking Factor	1.7	1.7
Core Average Burnup (GWD/MTU)	34.3	40
Fuel Decay (days)	1	60
Gap Release Fractions	8% I-131, 10% Kr-85, 5% Other Noble Gas, 5% Other Halogens, 12% Alkali Metals	8% I-131, 10% Kr-85, 5% Other Noble Gas, 5% Other Halogens, 12% Alkali Metals
Core Inventory After Decay (Ci/MWth)	I-131 2.52E+4 I-132 3.20E+4 I-133 2.53E+4 I-134 1.30E-3 I-135 4.09E+3 Kr-85 3.32E+2 Kr-85m 1.68E+2 Kr-87 2.74E-2 Kr-88 5.21E+1 Xe-133 5.13E+4 Xe-135 1.38E+4	I-131 1.60E+2 I-132 1.14E-1 I-133 8.10E-17 I-134 0 I-135 0 Kr-85 3.62E+2 Kr-85m 0 Kr-87 0 Kr-88 0 Xe-133 2.36E+1 Xe-135 0
Iodine Form of Gap Release	99.85% Elemental 0.15% Organic	99.85% Elemental 0.15% Organic
Water Decontamination Factors	500 Elemental 1 Organic 1 Noble Gas Infinite Particulates	500 Elemental 1 Organic 1 Noble Gas Infinite Particulates
Duration of Release to Environment (hours)	2	2
Standby Gas Treatment System Credit	None	None
Secondary Containment Volume Credit	None	None
Offsite and Control Room Breathing Rate (m^3/s)	3.47E-4	3.47E-4
Control Room Free Volume (ft^3)	324,100	324,100
Control Room Filtration Credit	None	None
Time After Accident to Control Room Isolation (minutes)	None	31

Control Room Flow Prior to Isolation (cfm)	5,060 unfiltered infiltration 5,060 exfiltration	5,060 unfiltered infiltration 5,060 exfiltration
Control Room Flow After Isolation (cfm)	5,060 unfiltered infiltration 5,060 exfiltration	0 infiltration 0 exfiltration
Control Room Occupancy Factors	1.0, 0 to 1 day 0.6, 1 to 4 days 0.4, 4 to 30 days	1.0, 0 to 1 day 0.6, 1 to 4 days 0.4, 4 to 30 days
0 to 2 hour Atmospheric Dispersion Factors (s/m^3)	Control Room 4.61E-4 EAB 1.8E-4 LPZ 4.2E-5	Control Room 6.25E-3 EAB 2.46E-4 LPZ 5.62E-5
Dose Conversion Factors	Federal Guidance Reports 11 and 12	Federal Guidance Reports 11 and 12

Spent Fuel Pool – Neutron Absorbing Material

This section describes the actions that will be taken to maintain and monitor the neutron absorbing materials in the SFP during decommissioning. This section is based on a request for additional information at Vermont Yankee Nuclear Power Station (Reference 15) and Exelon's response to NRC Generic Letter 2016-01 "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools" (Reference 16).

There are three types of spent fuel storage racks in use at CPS. Two types of high-density racks utilizing a neutron poison are installed in the fuel building spent fuel storage pool. The fuel cask storage pool (FCSP) may also be used to hold up to two high-density storage racks, as needed, to extend core offload capacity. The cast aluminum racks in the upper containment fuel storage area will no longer be used once the reactor is permanently defueled. The SFP contains a mix of fuel storage racks with Boral® Neutron Absorbing Material (NAM) and Metamic NAM (i.e., both built into stainless steel rack structures). No single rack contains more than one type of NAM. The two spent fuel racks reserved for the FCSP contain Boral® NAM.

The SFP contains 26 storage racks to store a maximum of 3,796 fuel assemblies. The FCSP may be utilized on an as-need basis (to extend core offload capacity) for storage of up to 2 storage racks to store a maximum of 264 fuel assemblies. CPS is licensed to store 3,796 fuel assemblies in the spent fuel pool and an additional 363 in the fuel cask storage pool, as needed. Ten of the 22 original-equipment Boral® racks remain in the SFP today. Sixteen Metamic SFP racks were added to expand spent fuel pool storage capacity in 2005 and 2007. The design of the spent fuel storage racks provides for a subcritical multiplication factor Keff of less than or equal to 0.95 for normal and abnormal storage conditions.

CPS utilizes periodic coupon measurements to monitor the ability of the Metamic 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. Metamic 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.

One Metamic coupon is removed after 2, 4, 8, 12, 20, 28, and 36 years. Coupons are also subject to neutron attenuation measurements after 4, 12, and 20 years to confirm the

attenuation capabilities. This periodicity is sufficient to provide an indication of degradation of the Metamic material prior to reaching an impact of more than 5 percent of the subcriticality margin. This is based on the industry experience with Metamic that has not shown any mechanism that leads to the loss of the boron from the Metamic material. The NRC accepted this coupon frequency and sampling size in Reference 17. EGC also monitors industry experience with the Metamic NAM through operating experience reviews and through industry group participation (e.g., NEI, EPRI).

CPS currently has no coupon or other monitoring program for the Boral® NAM in the SFP. EGC monitors Boral® operating experience from its own fleet of plants with Boral® racks and also through participation in industry groups such as INPO, EPRI Neutron Absorber Users Group, and the NEI Criticality Task Force. There have been no industry reports that have identified a concern with respect to the Boral® having a loss of B-10 areal density.

EPRI Report 1025204, "Strategy for Managing the Long Term Use of Boral® in Spent Fuel Storage Pools," July 2012 (Reference 18), contains the following conclusions:

> Possible BORAL® performance issues can be divided into two main classes:
>
> A) Safety (criticality safety) relative to the functionality of the B-10 in the core material; and,
>
> B) Operational as characterized by blistering of the BORAL® surface where the aluminum cover material (cladding) separates from the core Boron containing material thus potentially effecting fuel handling.

BORAL® has successfully met all the criticality safety performance requirements for over 25 years of service as demonstrated by the following considerations:

- There have been no surveillance data or observed cases where there has been significant loss or redistribution of B-10 from BORAL®.
- No mechanisms have been identified or observed that would lead to severe degradation of the BORAL® core material.
- No mechanisms have been identified that would lead to a sudden loss or reconfiguration of the BORAL® core material.

Clinton Power Station is continuing with its current programs and processes described above to monitor the fuel in the SFP since CPS does not have a renewed operating license. These programs and processes will continue as long as spent fuel is stored in the SFP. All spent fuel is scheduled to be placed into dry casks and stored on the Independent Spent Fuel Storage Installation (ISFSI) by 2023. This will be reflected in the Post-Shutdown Decommissioning Activities Report which is planned to be submitted in early 2017. Therefore, the current program and processes are more than adequate to manage aging of equipment relied upon to safely store fuel within the SFP.

Detailed Discussion

The following table identifies each FOL and TS section that is being changed, the proposed changes, and the basis for each change. Changes to the FOL are listed first followed by the TS.

Attachment 2 provides the marked-up version of the CPS FOL and TS to establish the changes. The TS 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. The following administrative changes are not shown in the marked-up (Attachment 2) FOL and TS pages, because they do not affect the technical content of the FOL or TSs:

- Reformatting (margins, font, tabs, etc.) content to create a continuous electronic file; and
- Renumbering of pages, where appropriate, to condense and reduce the number of pages.

Proposed changes to the TS Bases are provided for information in Attachment 3. The TS Bases Sections that are deleted in their entirety are not included in Attachment 3. Upon approval of this amendment, changes to the Bases will be incorporated in accordance with TS 5.5.11, the TS Bases Control Program.

License Finding 1.B.	
<u>Current License Finding 1.B.</u> Construction of the Clinton Power Station, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-137 and the application, as amended, the provisions of the Act and the regulations of the Commission;	<u>Proposed License Finding 1.B.</u> Deleted
Basis	
This License Finding is proposed for deletion in its entirety. Decommissioning of CPS is not dependent on the regulations that govern construction of the facility.	

License Finding 1.C.	
<u>Current License Finding 1.C.</u> The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);	<u>Proposed License Finding 1.C.</u> The facility ~~will operate~~ ***is prohibited from operating the reactor*** 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 revised to reflect the change from an operating license to being prohibited from operating the reactor. Once CPS has permanently ceased operation and certified that fuel has been 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)	
<u>Current License Condition 2.B.(1)</u> Exelon Generation Company, pursuant to section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in Harp Township, DeWitt County, Illinois, in accordance with the procedures and limitations set forth in this license;	<u>Proposed License Condition 2.B.(1)</u> Exelon Generation Company, pursuant to section 103 of the Act and 10 CFR Part 50, to possess~~,~~ ***and*** use ~~and operate~~ the facility at the designated location in Harp Township, DeWitt County, Illinois, in accordance with the procedures and limitations set forth in this license;
Basis	
This License Condition is revised 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.(3)	
Current License Condition 2.B.(3) Exelon Generation Company, pursuant to the Act and 10 CFR Part 70, to receive, possess and to 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 Final Safety Analysis Report, as supplemented and amended;	Proposed License Condition 2.B.(3) Exelon Generation Company, pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess ~~and to 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 Final Safety Analysis Report, as supplemented and amended;
Basis	
This License Condition is revised to reflect the change from an operating license to being prohibited from operating the reactor and that special nuclear material (SNM) can no longer be used as reactor fuel. The proposed change removes the authorization for receipt and use of SNM as reactor fuel and eliminates the reference to use of the SNM for reactor operations. The proposed change limits the possession of SNM to SNM “that was used” as reactor fuel. Once CPS has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) will prohibit operation of the CPS reactor.	

License Condition 2.B.(4)	
Current License Condition 2.B.(4) Exelon Generation Company, pursuant to the Act and to 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material 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.(4) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30,40, and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources ***that were used*** for reactor startup, sealed sources ***that were used*** for reactor instrumentation ***calibration*** and ***are used in*** radiation monitoring equipment calibration, and as fission detectors in amounts as required;
Basis	
This License Condition is revised regarding receipt and use of sealed neutron sources for reactor startup. This License Condition is revised to reflect authorization only for continued possession of those sources used for reactor startups. The Part 50 license will no longer authorize operation of the facility pursuant to 10 CFR 50.82(a)(2), this License Condition is being revised to no longer authorize receipt and use of byproduct, source, and SNM for reactor startup or for calibration of reactor instrumentation. The License Condition will continue to allow use of sealed sources in radiation monitoring equipment calibration and as fission detectors.	

License Condition 2.B.(6)	
Current License Condition 2.B.(6) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation; and	Proposed License Condition 2.B.(6) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~ ***that were*** produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation; and
Basis	
This License Condition is revised regarding the possession of byproduct and SNM, but not the separation of material that may be produced by operation 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 revised to allow	

possession of byproduct and SNM that were produced during operation of the reactor, but not the separation of material that was produced by operations of the reactor.

License Condition 2.C.(1)	
Current License Condition 2.C.(1) Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3473 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 CPS has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the CPS reactor since the certifications described therein will have been submitted.	

License Condition 2.C.(5)	
Current License Condition 2.C.(5) Exelon Generation Company shall store new fuel assemblies in accordance with the requirements specified in Attachment 2. Attachment 2 is hereby incorporated into this license. In conjunction with deletion of License Condition 2.C.(5), Attachment 2 to NPF-62 "NEW FUEL STORAGE," is also being deleted.	Proposed License Condition 2.C.(5) Deleted
Basis	
This License Condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once CPS has permanently ceased operation and certified that fuel has been removed from the reactor, pursuant to 10 CFR 50.82(a)(2) the 10 CFR 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel.	

License Condition 2.C.(23)	
Current License Condition 2.C.(23) Upon implementation of Amendment No. 178 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.3.5, in accordance with TS 5.5.15.c.(i), the assessment of CRE habitability as required by Specification 5.5.15.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.15.d, shall be considered met. Following implementation: (a) The first performance of SR 3.7.3.5, in accordance with Specification 5.5.15.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from November 16, 2004, the date of the most recent successful tracer	Proposed License Condition 2.C.(23) Deleted

<table>
<tr><td>gas test, as stated in the February 8, 2005 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.15.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from November 16, 2004, the date of the most recent successful tracer gas test, as stated in the February 8, 2005 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.15.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test, or within 6 months if not performed previously.</td><td></td></tr>
<tr><td colspan="2">Basis</td></tr>
<tr><td colspan="2">This License Condition supported implementation of the TS changes associated with the Control Room Envelope (CRE) Habitability program by requiring initial performance of the required surveillance SR 3.7.3.5, the first performance of the required periodic assessment of CRE habitability, and the first performance of the periodic measurement of CRE pressure within the specified frequency. These surveillances and assessments were completed in accordance with the schedule specified in the License Condition.

This License Condition is proposed for elimination. Following 60 days of decay post-shutdown, the FHA analysis (IP-F-0179) shows that the dose consequences to occupants in the control room are acceptable without relying on the main control room envelope to remain functional during and following a FHA. The dose at the control room would be 4.61 rem, which is less than the 10 CFR 50.67 dose limit of 5 rem. The calculation did not credit ventilation isolation or filtered recirculation to limit inleakage. As such, a Control Room Envelope (CRE) Habitability Program is not required to provide airborne radiological protection for the control room operators. This submittal also proposes to remove TS 3.7.3 for control room ventilation system and TS 5.5.15 for the CRE Habitability Program. Since TS 3.7.3 and TS 5.5.15 are no longer necessary, this License Condition is no longer needed; therefore, the proposed deletion of this License Condition is acceptable.</td></tr>
</table>

<table>
<tr><td colspan="2">License Condition 2.D.</td></tr>
<tr><td><u>Current License Condition 2.D.</u>

The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a) an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storage area; (b) an exemption from the requirement of 10 CFR Part 50, Appendix J - Option B, paragraph III.B, exempting</td><td><u>Proposed License Condition 2.D.</u>

The facility requires ~~exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a)~~ an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storage area.~~; (b) an exemption from the requirement of 10 CFR Part 50, Appendix J - Option B, paragraph III.B, exempting~~</td></tr>
</table>

the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for local leak rate tests (Section 6.2.6 of SSER 6); and (c) an exemption from the requirements of paragraph III. B of Option B of 10 CFR Part 50, Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests (SER supporting Amendment 62 to Facility Operating License No. NPF-62). The special circumstances regarding each exemption, except for item (a) above, are identified in the referenced section of the safety evaluation report and the supplements thereto. An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC Material License No. SNM-1886, issued November 27, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Exelon Generation Company is hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license. These exemptions 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. The exemptions in items (b) and (c) above are granted pursuant to 10 CFR 50.12. 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.	~~the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for local leak rate tests (Section 6.2.6 of SSER 6); and (c) an exemption from the requirements of paragraph III. B of Option B of 10 CFR Part 50, Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests (SER supporting Amendment 62 to Facility Operating License No. NPF-62). The special circumstances regarding each exemption, except for item (a) above, are identified in the referenced section of the safety evaluation report and the supplements thereto.~~ An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC Material License No. SNM-1886, issued November 27, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Exelon Generation Company is hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license. These exemptions 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. ~~The exemptions in items (b) and (c) above are granted pursuant to 10 CFR 50.12.~~ With ~~these~~ ***this*** exemption~~s~~, 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 documents specific exemptions from 10 CFR Part 50 and 10 CFR Part 70 as approved by the NRC. Once CPS has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) will prohibit operation of the CPS reactor. Therefore, CPS will no longer be performing leak testing of the main steam isolation valves and valve 1E51-F374 associated with containment penetration 1MC-44. Since these leak tests will no longer be performed, there is no longer a need for the specified exemption and the details of these exemptions are being deleted from this License Condition. The exemption to 10 CFR Part 70 remains unchanged.	

License Condition 2.F.	
<u>Current License Condition 2.F.</u> Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety	<u>Proposed License Condition 2.F.</u> Deleted

<table>
<tr><td>Analysis Report as amended, for the Clinton Power Station, Unit No. 1, and as approved in the Safety Evaluation Report (NUREG-0853) dated February 1982 and Supplement Nos. 1 thru 8 thereto subject to the following provision:

Exelon Generation Company 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.</td><td></td></tr>
<tr><td colspan="2">Basis</td></tr>
<tr><td colspan="2">This License Condition is proposed for deletion to reflect the permanently defueled condition of the facility. Once CPS has permanently ceased operation and certified that fuel has been removed from the reactor, the fire protection program will be revised to take into account the facility conditions and activities during decommissioning. CPS 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 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 CPS. 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. However, 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.</td></tr>
</table>

<table>
<tr><td colspan="2">License Condition 2.I.</td></tr>
<tr><td><u>Current License Condition 2.I.</u>

This license is effective as of the date of issuance and shall expire at midnight on September 29, 2026.</td><td><u>Proposed License Condition 2.I.</u>

This license is effective as of the date of issuance and ~~shall expire at midnight on September 29, 2026~~is effective until the Commission notifies the licensee in writing that the license is terminated.</td></tr>
<tr><td colspan="2">Basis</td></tr>
<tr><td colspan="2">This License Condition is revised to reflect the permanently defueled condition of the facility. Once CPS has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the CPS reactor since the certifications required by 10 CFR 50.82(a)(1) will have been submitted. This License Condition is being revised in accordance with 10 CFR 50.51(b) in that the license authorizes ownership and possession by Exelon until the Commission notifies the licensee in writing that the license is terminated.</td></tr>
</table>

<table>
<tr><td colspan="2">Facility Operating License No. NPF-62 Attachments and Appendix C</td></tr>
<tr><td><u>Current License</u>

Attachment 1 – Previously deleted
Attachment 2 – New Fuel Storage (See 2.C.(5) above)
Appendix C - Previously deleted</td><td><u>Proposed License</u>

[Pages removed]</td></tr>
<tr><td colspan="2">Basis</td></tr>
<tr><td colspan="2">This proposed change is to remove deleted pages for pagination and is administrative in nature.</td></tr>
</table>

TS Section 1.1 – Definitions	
TS 1.1, "Definitions," provides defined terms that are applicable throughout the TS and TS Bases. New definitions for Certified Fuel Handler and Non-Certified Operator were previously proposed for addition (reference 2 and 3). The definitions identified below for deletion pertain to an operating reactor and, therefore, they would no longer apply to the permanently defueled reactor.	
Proposed Definitions Added	**Basis**
CERTIFIED FUEL HANDLER	The definition for Certified Fuel Handler was proposed for addition in the LAR proposing changes to TS Sections 1.1 and 5.0 (Reference 2) currently under NRC review.
NON-CERTIFIED OPERATOR	The definition for Non-Certified Operator was proposed for addition in the supplemented LAR proposing changes to TS Sections 1.1 and 5.0 (Reference 3) currently under NRC review.
Proposed Definitions Deleted	**Basis**
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term is only meaningful to a reactor authorized to operate.
CHANNEL CALIBRATION	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible.
CHANNEL CHECK	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible.
CHANNEL FUNCTIONAL TEST	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible.
CORE ALTERATION	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term has no meaning when there is no fuel in the reactor core.
CORE OPERATING LIMITS REPORT (COLR)	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification and Specification 5.6.5 that requires the COLR is also proposed for elimination.
DOSE EQUIVALENT I-131	This definition is not proposed for inclusion in the PDTS since this term is not used in any PDTS specification. This term is used to express dose from a mixture of iodine isotopes created in an operating core and contained in plant primary or secondary coolant. The value of Dose Equivalent I-131 is used for dose analysis of accidents involving primary coolant releases. Those accident conditions will no longer apply to the permanently shut down and defueled facility.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. ECCS will have no function in the permanently shut down and defueled facility.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. Reactor Recirculation Pumps will have no function in the permanently shut down and defueled facility.
ISOLATION SYSTEM RESPONSE TIME	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible.
LEAKAGE	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. None of the SSCs from or into which leakage is monitored are credited in the analysis of the accidents that remain credible.
LINEAR HEAT GENERATION RATE (LHGR)	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term is only meaningful to a reactor authorized to operate.
LOGIC SYSTEM FUNCTIONAL TEST	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There are no logic systems credited in the analysis of the accidents that remain credible.
MINIMUM CRITICAL POWER RATIO (MCPR)	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term is only meaningful to a reactor authorized to operate.
MODE In conjunction with deletion of the term "Mode," TS Table 1.1-1, "MODES," is also being deleted.	This definition, including TS Table 1.1-1, is not proposed for inclusion in the PDTS since operating Modes are not used in any PDTS specification. Modes are defined for operating or refueling conditions. This term does not apply to a facility in the permanently defueled condition.
RATED THERMAL POWER (RTP)	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term is only meaningful to a reactor authorized to operate.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. The RPS will have no function in the permanently shut down and defueled facility.
SHUTDOWN MARGIN (SDM)	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term does not apply to a facility in the permanently defueled condition.
STAGGERED TEST BASIS	This definition is not proposed for inclusion in the PDTS since the

	term is not used in any PDTS specification. This definition applies to the performance of surveillance tests on systems with multiple subsystems or channels. There are no surveillance requirements in the PDTS for operating systems.
THERMAL POWER	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term is only meaningful to a reactor authorized to operate.
TURBINE BYPASS SYSTEM RESPONSE TIME	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. The main turbine bypass system has no function in the permanently defueled condition.

TS SECTION 1.2, LOGIC CONNECTORS
TS 1.2, "Logical Connectors," contains an explanation of the logical connectors used to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies throughout TS. Logic Connectors are no longer used in the PDTS. This section is proposed for deletion in its entirety.

TS SECTION 1.3, COMPLETION TIMES	
BACKGROUND	This is proposed for revision to remove reference to "operation of the unit" and replace it with reference to "handling and storage of spent nuclear fuel." Proposed changes are shown in Attachment 2.
DESCRIPTION	This explanation is proposed for revision to remove discussion of Modes that will not exist in a permanently defueled facility. Reference to "inoperable equipment" is proposed for elimination because the specifications remaining in PDTS apply only to a variable (SFP Level). The term "unit" is typically associated with an operating reactor and is revised with the term "facility." This administrative change more appropriately represents the permanently shutdown and defueled condition. Proposed changes are shown in Attachment 2.
EXAMPLES	The examples in this section are proposed for deletion. The examples are no longer necessary because they describe examples of Completion Times that do not remain in the PDTS. The Action that remains in the PDTS must be completed "Immediately" which is retained in PDTS section 1.3.

TS SECTION 1.4, FREQUENCY	
DESCRIPTION	This explanation is proposed for revision to remove discussion of surveillance performance situations that do not exist in the PDTS. Proposed changes are shown in Attachment 2.
EXAMPLES	This section is proposed for revision to remove discussion of surveillance performance situations that do not exist in the PDTS, and to explicitly address those that do exist. An administrative change in this section replaces the term "unit" with the term "facility." Proposed changes are shown in Attachment 2.

TS SECTION 2.0, SAFETY LIMITS (SLs)
All TS in Section 2.0, Safety Limits (SLs) are being proposed for deletion. The safety limits do not apply to a reactor that is in a permanently defueled condition. TS 2.1, "Safety Limits" (SLs), contains two separate specifications: TS 2.1.1, Reactor Core SLs; and TS 2.1.2, Reactor Coolant System Pressure SL The restrictions of the SLs promulgated in TS 2.1.1 are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. TS 2.1.1 is applicable in Modes 1, 2, 3, 4, and 5. TS 2.1.2 provides requirements on parameters to protect the integrity of the RCS against overpressure. TS 2.1.2 is applicable in Modes 1, 2, 3, 4, and 5. TS 2.2, SL Violations, directs actions to be taken if a safety limit specified in TS 2.1 is violated. TS 2.2 is applicable commensurate with the applicable Modes of the respective safety limits specified in TS 2.1. The above TS contain limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the facility only when the reactor is in Modes 1 through 5. However, 10 CFR 50.82(a)(2) will prohibit operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which address only these specific process variables will no longer be applicable. Based on the above, the proposed deletion of the TS related to these Safety Limits is acceptable.

TS SECTION 3.0, LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	
TS 3.0, "Limiting Condition for Operation (LCO) Applicability," consist of LCOs 3.0.1 through LCO 3.0.7. These LCOs establish the general requirements applicable to all TSs and apply at all times, unless otherwise stated. The proposed revisions to these LCOs are shown in Attachment 2.	
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual TS as the requirement for when the LCO is required to be met (i.e., when the unit is in the Mode or other specified condition of the Applicability statement of each Specification). LCO 3.0.1 is being retained in the PDTS with the proposed revisions shown in Attachment 2. Because 10 CFR 50.82(a)(2) will prohibit operation of the plant or placing fuel in the reactor vessel, the reference to operating Modes will no longer be relevant and is therefore being deleted. Additionally, reference to LCO 3.0.7 is also being deleted to conform to deletion of this LCO as discussed below.
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. LCO 3.0.2 is proposed for revision to remove references to LCO 3.0.5 and LCO 3.0.6 to conform to deletion of these LCOs as discussed below.
LCO 3.0.3	LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and the associated actions are not met, an associated action is not provided, or if directed by the associated action. LCO 3.0.3 is only applicable in Modes 1, 2, and 3. LCO 3.0.3 does not apply with the reactor defueled. LCO 3.0.3 is being proposed for deletion in its entirety.

	Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, LCO 3.0.3 will no longer be applicable. Therefore, the proposed deletion of LCO 3.0.3 is acceptable.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in Modes or other specified conditions in the Applicability when an LCO is not met. LCO 3.0.4 is proposed for deletion because all actions in the PDTS have a completion time of "Immediately." This makes LCO 3.0.4 unnecessary.
LCO 3.0.5	LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. LCO 3.0.5 is proposed for deletion, because there are no LCOs for equipment to be operable or in operation in the PDTS.
LCO 3.0.6	LCO 3.0.6 addresses the actions required for a supported system when the support system LCO is not met. LCO 3.0.6 is proposed for deletion since there are no LCOs for equipment to be operable or in operation in the PDTS.
LCO 3.0.7	LCO 3.0.7 pertains to certain special tests and operations required to be performed at various times over the life of the unit. LCO 3.0.7 is proposed for deletion since special tests and operations are not applicable to a permanently defueled facility. This aligns with the proposed deletion of TS LCO Section 3.10.

TS SECTION 3.0, SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	
TS 3.0, "Surveillance Requirement (SR) Applicability," consists of SRs 3.0.1 through SR 3.0.4. These SRs establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. The proposed revisions to these LCOs are shown in Attachment 2.	
SR 3.0.1	SR 3.0.1 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. SR 3.0.1 is proposed for revision to remove references to operating Modes and inoperable equipment since there are no LCOs for equipment to be operable or in operation in the PDTS. 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.
SR 3.0.2	SR 3.0.2 provides 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. SR 3.0.2 is proposed for revision to remove conditions for frequencies that do not exist in PDTS LCOs.

SR 3.0.4	SR 3.0.4 establishes requirements that all applicable SRs must be met before entry into an operational Mode or other specified condition in the applicability. SR 3.0.4 is being modified, such that, the surveillance requirement in TS 3.7.7 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. Because 10 CFR 50.82(a)(2) will prohibit operation of the plant or placing fuel in the reactor vessel, the reference to operating Modes 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 Section 3.1, REACTIVITY CONTROL SYSTEMS	
TS Section 3.1 contains LCOs and SRs to assure and verify operability of reactivity control systems. Once CPS 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, in accordance with 10 CFR 50.82(a)(2). Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, reactivity control systems will not be required and these LCOs (and associated SRs) will not apply in a defueled condition. Therefore, TS Section 3.1 is proposed for deletion in its entirety.	
TS 3.1.1, SHUT DOWN MARGIN (SDM)	This specification defines the minimum shut down margin in the reactor core for Modes 1, 2, 3, 4, and 5. SDM was included in the Improved Technical Specifications (ITS) to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.1.1 is not proposed for inclusion in the PDTS since CPS will be permanently shut down and defueled. This TS does not provide protection for the cladding of fuel stored in the SFP.
TS 3.1.2, Reactivity Anomalies	This specification defines the required accuracy for measured vs. predicted rod density. This specification is applicable during Modes 1 and 2. Reactivity anomaly was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.1.2 is not included in the PDTS since CPS will be permanently shut down and defueled, and this TS does not provide protection for the cladding of fuel stored in the SFP.
TS 3.1.3, Control Rod OPERABILITY	This specification defines the operability requirements for the control rods. This specification is applicable during Modes 1 and 2. Control rod OPERABILITY was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.1.3 is not proposed for inclusion in the PDTS, since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. At that time, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel.
TS 3.1.4, Control Rod Scram Times	This specification defines the control rod scram times. This specification is applicable during Modes 1 and 2. Control rod scram times were

	included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.1.4, including Table 3.1.4-1, "Control Rod Scram Times," is not proposed for inclusion in the PDTS, since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. At that time, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel.
TS 3.1.5, Control Rod Scram Accumulators	This specification defines the operability requirements for the control rod scram accumulators. This specification is applicable during Modes 1 and 2. Control rod scram accumulators were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.1.5 is not proposed for inclusion in the PDTS, since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. At that time, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel.
TS 3.1.6. Control Rod Pattern	This Specification defines the control rod sequences to assure that the control rod patterns are consistent with the assumptions of the Control Rod Drop Accident analyses. This specification is applicable during Modes 1 and 2 with THERMAL POWER $\leq$ 16.7% RTP. Rod pattern control was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.1.6 is not proposed for inclusion in the PDTS, since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. At that time, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel.
TS 3.1.7, Standby Liquid Control (SLC) System	This specification defines the operability requirements for the SLC System. This specification is applicable during Modes 1, 2, and 3. The SLC System was included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.1.7, including Figure 3.1.7-1, "Weight Percent Sodium Pentaborate Solution Concentration/Net Tank Volume Requirements," is not proposed for inclusion in the PDTS, since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. At that time, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel.
TS 3.1.8, Scram Discharge Volume (SDV) Vent and Drain Valves	This specification defines the operability requirements for the SDV vent and drain valves. This specification is applicable during Modes 1 and 2. The SDV vent and drain valves were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.1.8 is not proposed for inclusion in the PDTS, since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. At that time, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel.

<table>
<tr><th colspan="2">TS SECTION 3.2, POWER DISTRIBUTION LIMITS</th></tr>
<tr><td colspan="2">TS Section 3.2 contains LCOs and SRs to ensure that power distribution limits are met. Once CPS 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, in accordance with 10 CFR 50.82(a)(2). Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, power distribution limits will not be required and these LCOs (and associated SRs) will not apply in a defueled condition. Therefore, TS Section 3.2 is proposed for deletion in its entirety.</td></tr>
<tr><td>TS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)</td><td>This specification 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 when THERMAL POWER $\geq$ 21.6% RTP. The APLHGR was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS 3.2.1 is not proposed for inclusion 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. This TS does not provide protection for the cladding of fuel stored in the SFP.</td></tr>
<tr><td>TS 3.2.2, MINIMUM CRITICAL POWER RATIO (MCPR)</td><td>This specification defines limits for the MCPR to ensure that no fuel damage results during abnormal operational transients. This specification is applicable when THERMAL POWER $\geq$ 21.6% RTP. The MCPR was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS 3.2.2 is not included 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. This TS does not provide protection for the cladding of fuel stored in the SFP.</td></tr>
<tr><td>TS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)</td><td>This specification 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 when THERMAL POWER $\geq$ 21.6% RTP. The LHGR was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

TS 3.2.3 is not included 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. This TS does not provide protection for the cladding of fuel stored in the SFP.</td></tr>
</table>

<table>
<tr><th>TS SECTION 3.3, INSTRUMENTATION</th></tr>
<tr><td>TS Section 3.3 contains LCOs and SRs to assure and verify operability of instrumentation systems. Once CPS 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, in</td></tr>
</table>

accordance with 10 CFR 50.82(a)(2). 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 ventilation (CRV) systems. Following 60 days of decay post-shutdown, the dose consequences to occupants in the control room are acceptable without relying on the control room systems to remain functional during and following a FHA. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the instrumentation addressed in TS Section 3.3 will not be required and these LCOs (and associated SRs) will not apply in a defueled condition. Therefore, TS Section 3.3 is proposed for deletion in its entirety.	
TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation	This specification provides the operability requirements for the RPS instrumentation specified in TS Table 3.3.1.1-1. The RPS Instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). Functions not specifically credited in the accident analysis are retained for the RPS as required by the NRC approved licensing basis. The Scram Discharge Volume water level – high function was retained in the ITS to ensure the RPS remained operable. The Drywell Pressure – High, Intermediate Range Monitor – Inop, Average Power Range Monitor – Inop, Reactor Mode Switch – Shut down Position and the Manual Scram functions were retained in the ITS for the RPS as required by the NRC approved licensing basis. TS 3.3.1.1, including TS Table 3.3.1.1-1, is not proposed for inclusion 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. Thus, there will no longer be a need for RPS instrumentation to protect the core.
TS 3.3.1.2, Source Range Monitor (SRM) Instrumentation	This specification provides the operability requirements for the SRM instrumentation specified in TS Table 3.3.1.2-1. The SRM Instrumentation was retained in the ITS, because they provided the only on-scale monitoring of neutron flux levels during startup and refueling. TS 3.3.1.2, including TS Table 3.3.1.2-1, is not included 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. Thus, there will no longer be a need for SRM instrumentation to monitor flux levels during startup and refueling.
TS 3.3.1.3, Oscillation Power Range Monitor (OPRM) Instrumentation	This specification provides the operability requirements for the Oscillation Power Range Monitor (OPRM) Instrumentation to detect and suppress neutron flux oscillations in the event of thermal-hydraulic instability. This specification is applicable when THERMAL POWER ≥ 21.6% RTP. The OPRM Instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.1.3 is not included 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. Thus, there will no longer be a need for monitoring to detect events of thermal-hydraulic instability.
TS 3.3.2.1, Control Rod Block Instrumentation	This specification provides the operability requirements for the Control Rod Block instrumentation specified in TS Table 3.3.2.1-1. The Control Rod Block Instrumentation was included in the ITS to satisfy Criterion 3

	of 10 CFR 50.36(c)(2)(ii). TS 3.3.2.1, including TS Table 3.3.2.1-1, is not included 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. Thus, there will no longer be a need for control rods to control core reactivity.
TS 3.3.3.1, Post Accident Monitoring (PAM) Instrumentation	This specification provides the operability requirements for the PAM instrumentation specified in TS Table 3.3.3.1-1. This specification is applicable during Modes 1 and 2. The PAM instrumentation that satisfied the definition of Type A in Regulatory Guide 1.97 was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). PAM instrumentation that was Category 1, not Type A, instrumentation were retain in the ITS, because they were intended to assist operators in minimizing the consequences of accidents. These Category 1 variables were considered important for reducing public risk. TS 3.3.3.1, including TS Table 3.3.3.1-1, is not included 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. The instrumentation specified in TS Table 3.3.3.1-1 would not provide information to operators during a FHA in the SFP. Thus, there will no longer be a need for PAM instrumentation.
TS 3.3.3.2, Remote Shutdown System	This specification provides the operability requirements for the instrumentation and controls necessary to place and maintain the plant in Mode 3 from a location other than the control room. This specification is applicable during Modes 1 and 2. The Remote Shutdown System is considered an important contributor to reducing the risk of accidents; as such, it has been retained in the TS as indicated in the NRC Policy Statement. TS 3.3.3.2 is not included 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. Thus, there will no longer be a need for the Remote Shutdown System to shutdown the reactor.
TS 3.3.4.1, End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	This specification provides the operability requirements for the EOC-RPT instrumentation. This specification is applicable when THERMAL POWER $\geq$ 33.3% RTP with any recirculation pump in fast speed. This instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.4.1 is not included 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. A pressurization transient will not be a credible event once the plant is in the permanently defueled condition. Thus, there will no longer be a need for EOC-RPT instrumentation.
TS 3.3.4.2, Anticipated Transient Without Scram	This specification provides the operability requirements for the ATWS-RPT instrumentation. This specification is applicable in Mode 1. This

Recirculation Pump Trip (ATWS-RPT) Instrumentation	instrumentation was included in the ITS to meet the requirements of 10CFR 50.62(c)(5) and satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.3.4.1 is not included 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. An ATWS will not be a credible event once the plant is in the permanently defueled condition. Thus, there will no longer be a need for ATWS-RPT instrumentation.
TS 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation	This specification provides the operability requirements for the ECCS instrumentation specified in TS Table 3.3.5.1-1. The ECCS instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). The Reactor Vessel Water Level – high, Level 8 function was retained in the TS, because it was a potentially significant contributor to risk. In addition, the manual initiation functions for the High Pressure Core Spray (HPCS) System, the Low Pressure Core Spray (LPCS) System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS), were retained as required by the NRC in the plant licensing basis. TS 3.3.5.1, including TS Table 3.3.5.1-1, is not included 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. Thus, there will no longer be a need for ECCS instrumentation.
TS 3.3.5.2, Reactor Core Isolation Cooling (RCIC) System Instrumentation	This specification provides the operability requirements for the RCIC instrumentation specified in TS Table 3.3.5.2-1. This specification is applicable in Mode 1 and in Mode 2 and 3 with reactor steam dome pressure > 150psig. The RCIC instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). The manual initiation function was retained in the ITS for the RCIC function as required by the NRC in the plant licensing basis. TS 3.3.5.2, including TS Table 3.3.5.2-1, is not included 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. Thus, there will no longer be a need for RCIC instrumentation.
TS 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation	This specification provides the operability requirements for the Primary Containment and Drywell Isolation instrumentation specified in TS Table 3.3.6.1-1. The Primary Containment and Drywell isolation instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.6.1, including TS Table 3.3.6.1-1, is not included 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. The FHA is the only credible DBA possible in the permanently defueled condition. The FHA analysis does not rely on primary containment to mitigate the consequences of the FHA. Therefore, this specification may be deleted.

TS 3.3.6.2, Secondary Containment Isolation Instrumentation	This specification provides the operability requirements for the secondary containment isolation instrumentation specified in TS Table 3.3.6.2-1. This specification is applicable in Mode 1, 2, and 3, or during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel in the primary or secondary containment or the fuel building. The secondary containment isolation instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.6.2, including TS Table 3.3.6.2-1, is not included 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. The requirement for Mode dependency will no longer apply. This specification will not be required after 24 hours of decay following shut down, because the nuclear fuel will no longer be considered to be "recently irradiated." In addition, the other condition requiring the operability of specific functions of the secondary containment isolation instrumentation (operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. Thus, there will no longer be a need for the secondary containment isolation instrumentation.
TS 3.3.6.3, Residual Heat Removal (RHR) Containment Spray System Instrumentation	This specification provides the operability requirements for the Residual Heat Removal (RHR) Containment Spray System Instrumentation specified in TS Table 3.3.6.3-1. This specification is applicable in Mode 1, 2, and 3. The RHR Containment Spray System Instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.6.3, including TS Table 3.3.6.3-1, is not included 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. The potential for containment pressure to exceed design limits will not occur; therefore, the instrumentation specified in TS Table 3.3.6.3-1 will no longer be needed to ensure that containment pressure is maintained within its limits following a loss of coolant accident (LOCA).
TS 3.3.6.4, Suppression Pool Makeup (SPMU) System Instrumentation	This specification provides the operability requirements for the Suppression Pool Makeup (SPMU) System Instrumentation specified in TS Table 3.3.6.4-1. This specification is applicable in Mode 1, 2, and 3. The SPMU System Instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.6.4, including TS Table 3.3.6.4-1, is not included 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. The potential for containment pressure and temperature to exceed design limits will not occur; therefore, the instrumentation specified in TS Table 3.3.6.4-1 will no longer be required to maintain the drywell horizontal vent coverage and adequate suppression pool heat sink volume.
TS 3.3.6.5, Relief and Low-	This specification provides the operability requirements for the Relief and Low-Low Set (LLS) Instrumentation. This specification is applicable in

Low Set (LLS) Instrumentation	Mode 1, 2, and 3. The Relief and LLS Instrumentation included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.6.5 is not included in the PDTS 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. The potential of an overpressurization of the nuclear steam system will not occur and the relief and LLS instrumentation will not be required to ensure that the containment loads remain within the primary containment design basis.
TS 3.3.7.1, Control Room Ventilation System Instrumentation	This specification provides the operability requirements for the Control Room Ventilation (CRV) System instrumentation. This specification is applicable in Mode 1, 2, and 3, or during operations with a potential for draining the reactor vessel, or during Core Alterations or during movement of irradiated fuel in the primary or secondary containment. This instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.7.1 is not included 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 following shut down. 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 ventilation (CRV) systems. The results of the evaluation indicated a 60 days decay time to meet the regulatory acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183 without credit for CRV systems, intake radiation monitors, and operator action in selecting control room intake with the lowest concentration of airborne activity. In addition, the other conditions requiring the operability of the CRV system instrumentation (Core Alterations or operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. TS LCO 3.3.7.1 is a support LCO that supports LCO 3.7.3. Therefore, after 60 days of decay following shutdown, the conditions requiring the CRV system instrumentation to be operable will no longer be applicable and the CRV system instrumentation will not be required to mitigate the consequences of the FHA, which is the remaining DBA of concern that will be possible in the permanently defueled condition. Thus, there will no longer be a need for the CRV system instrumentation
TS 3.3.8.1, Loss of Power (LOP) Instrumentation	This specification provides the operability requirements for the LOP instrumentation specified in TS Table 3.3.8.1-1. This specification is applicable during Modes 1, 2 and 3 and during a specified condition of "When the associated diesel generator (DG) is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown." This instrumentation was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.8.1, including TS Table 3.3.8.1-1, is not included 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. Thus, there will no longer be a need for this LCO during Modes 1, 2 or 3. With the

	elimination of TS LCO 3.7.3 and 3.7.4, the EDG no longer supports the systems that are required for movement of irradiated fuel. These TS are supported by TS LCO 3.8.9 and 3.8.10, which are supported by TS LCOs 3.8.1 and 3.8.2. This TS LCO also has a specified condition of "When the associated diesel generator (DG) is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown." Since LCO 3.8.2 is proposed for deletion (see discussion below), LCO 3.3.8.1 is proposed for deletion.
TS 3.3.8.2, Reactor Protection System (RPS) Electric Power Monitoring	This specification provides the operability requirements for the RPS electric power monitoring assemblies that isolate the RPS bus from the normal uninterruptible power supply (UPS) or alternate power supply in the event of overvoltage, undervoltage, or underfrequency. This specification is applicable in Modes 1, 2, and 3, or Modes 4 and 5 with any control rod withdrawn for a core cell containing one or more fuel assemblies. They were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.3.8.2 is not included 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. There will no longer be a need for RPS instrumentation to protect the core. TS 3.3.1.1 RPS Instrumentation is being eliminated as documented above. Thus, there will no longer be a need for RPS Electric Power Monitoring.

TS SECTION 3.4, REACTOR COOLANT SYSTEM (RCS)	
TS Section 3.4 contains LCOs and SRs that provide assurance of the integrity and safe operation of the RCS and the operation of the associated equipment. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) will not apply (or are no longer needed) in a defueled condition. Therefore, TS Section 3.4 is proposed for deletion in its entirety.	
TS 3.4.1, Recirculation Loops Operating	This specification provides the operability requirements for the Recirculation Loops. The Reactor Coolant Recirculation System provides forced coolant flow through the core to remove heat from the fuel. This specification is applicable in Mode 1 and 2. The Recirculation Loops Operating were included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.1 is not proposed for inclusion 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. Thus, there will no longer be a need for the Recirculation Loops.
TS 3.4.2. Flow Control Valves (FCVs)	This specification provides the operability requirements for the Reactor Coolant System flow control valves. A FCV in each operating recirculation loop must be operable to ensure that the assumptions of the design basis transient and accident analyses are satisfied. This specification is applicable in Mode 1 and 2. Flow control valves satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

	TS 3.4.2 is not included 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. Thus, there will no longer be a need for the flow control valves.
TS 3.4.3, Jet Pumps	This specification provides the operability requirements for the Jet pumps. The jet pumps are part of the Reactor Coolant Recirculation System and are designed to provide forced circulation through the core to remove heat from the fuel. This specification is applicable in Mode 1 and 2. The jet pumps were included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.3 is not included 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. Thus, there will no longer be a need for the jet pumps.
TS 3.4.4, Safety/Relief Valves (S/RVs)	This specification provides the operability requirements for the S/RVs. These valves provide overpressure protection to the reactor during operation. This specification is applicable in Mode 1, 2, and 3. The S/RVs were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.4.4 is not included 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. Thus, there will no longer be a need for the S/RVs.
TS 3.4.5, RCS Operational LEAKAGE	This specification provides the allowable leakage rates of reactor coolant from the RCS. The limits provided protection of the reactor coolant pressure boundary from degradation and the core from inadequate cooling. This specification is applicable in Mode 1, 2, and 3. Limits on RCS operational leakage were included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.5 is not included 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. Thus, there will no longer be a need for the limits on RCS operational leakage.
TS 3.4.6, RCS Pressure Isolation Valve (PIV) Leakage	This specification provides the allowable leakage rates for the Reactor Coolant System pressure isolation valves. This specification is applicable in Modes 1, 2, and 3 except during transition to or from shutdown cooling mode. The limits were intended to prevent overpressure failure of the low pressure portions of the systems connected to the RCS. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. Limits on RCS operational leakage were included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.6 is not included 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. Thus, there will no longer be a need for the limits on RCS PIV leakage.
TS 3.4.7, RCS Leakage Detection Instrumentation	This specification provides the operability requirements for the RCS leakage detection instrumentation. This specification is applicable in Mode 1, 2, and 3. They were included in the ITS to satisfy Criterion 1 of 10 CFR 50.36(c)(2)(ii). TS 3.4.7 is not included 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. Thus, there will no longer be a need for the RCS leakage detection instrumentation.
TS 3.4.8, RCS Specific Activity	This specification provides limits regarding RCS specific activity. This specification is applicable in Mode 1 or in Mode, 2 and 3 with any main steam line not isolated. These limits were included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.8 is not included 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. Thus, there will no longer be a need for the limits on RCS specific activity.
TS 3.4.9, Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown	This specification provides the operability requirements for the RHR shut down cooling system during hot shut down. This specification is applicable in Mode 3 with reactor steam dome pressure less than the RHR cut in permissive pressure. The RHR shut down cooling system was included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.4.9 is not included 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. Thus, the RHR shut down cooling mode of the RHR system will no longer be required.
TS 3.4.10, Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown	This specification provides the operability requirements for the RHR shut down cooling system during cold shut down. This specification is applicable in Mode 4. The RHR shut down cooling system was included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.4.10 is not included 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. Thus, the RHR shut down cooling mode of the RHR system will no longer be required.
TS 3.4.11, RCS Pressure and Temperature (P/T) Limits	This specification provides RCS P/T limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary. The reactor vessel is the component most subject to brittle failure; therefore, the LCO limits apply mainly to the vessel. Figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 contain composite P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH)

	testing. The RCS P/T limits provide a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G. Although the P/T limits were developed to provide guidance for operation during heatup or cooldown or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The P/T limits were included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.11, including Figures 3.4.11-1, 3.4.11-2, and 3.4.11-3, is not included in the PDTS since CPS will be permanently shut down and defueled. As such, the requirements of 10 CFR 50, Appendix G, no longer apply in such a condition because the reactor coolant pressure boundary will no longer be used as a fission product barrier when the reactor vessel is permanently defueled. Therefore, TS 3.4.3 is no longer needed and may be deleted.
TS 3.4.12, Reactor Steam Dome Pressure	This specification provides the limit on the reactor steam dome pressure. The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of DBAs and transients. This specification is applicable in Mode 1 and 2. The reactor steam dome pressure limit was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.4.12 is not included 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. The reactor vessel will remain depressurized; thus, there will no longer be a need for the reactor steam dome pressure limit.

TS SECTION 3.5, EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM TS Section 3.5 contains LCOs and SRs that provide assurance of the integrity and safe operation of the ECCS and RCIC System. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) will not apply (or are no longer needed) in a defueled condition. Therefore, TS Section 3.5 is proposed for deletion in its entirety.	
TS 3.5.1, ECCS - Operating	This specification provides operability requirements for the ECCS under operating conditions (i.e., Modes 1, 2, and 3). The ECCS network consists of the High Pressure Core Spray (HPCS) System, the Low Pressure Core Spray (LPCS) System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. The ECCS was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.5.1 is not included 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. Thus, there will no longer be a need for the ECCS.

TS 3.5.2, ECCS - Shutdown	This specification provides operability requirements for ECCS under Shutdown Conditions (i.e., Modes 4 and 5). These were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.5.2 is not included 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. Thus, there will no longer be a need for the low pressure ECCS subsystems.
TS 3.5.3, RCIC System	This specification provides operability requirements for RCIC system. This specification is applicable in Modes 1 or Modes 2, and 3 with reactor steam dome pressure > 150 psig. Its function was to respond to transient events by providing makeup coolant to the reactor. The RCIC System was included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.5.3 is not included 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. Thus, there will no longer be a need for the RCIC System.

TS SECTION 3.6, CONTAINMENT SYSTEMS	
TS Section 3.6 contains LCOs and SRs that provide assurance of the integrity and safe operation of the containment systems. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) will not apply (or are no longer needed) in a defueled condition. Therefore, TS Section 3.6 is proposed for deletion in its entirety.	
TS 3.6.1.1, Primary Containment	This specification provides operability requirements for the primary containment. Its function was to isolate and contain fission products released from the Reactor Primary System following a DBA and to confine the postulated release of radioactive material to within limits. This specification is applicable in Modes 1, 2 and 3. The primary containment was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.1 is not included 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. Thus, the requirements for primary containment are no longer applicable.
TS 3.6.1.2, Primary Containment Air Locks	This specification provides operability requirements for the primary containment air locks. This specification is applicable in Modes 1, 2, and 3, or during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel in the primary or secondary containment. The airlocks provide personnel access to the primary containment and provide containment isolation during the process of personnel entry and exit. The primary containment air locks were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.2 is not included 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. Thus, there will no longer be a need for the primary containment air locks.
TS 3.6.1.3, Primary Containment Isolation Valves (PCIVs)	This specification provides operability requirements for the PCIVs. Their function was to limit fission products released during and following a DBA to within limits. This specification is applicable in Modes 1, 2, 3, or during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel in the primary or secondary containment; and Modes 4 and 5 for RHR system Level 3 containment isolation. The PCIVs were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.3 is not included 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. Thus, there will no longer be a need for the PCIVs.
TS 3.6.1.4, Primary Containment Pressure	This specification provides a limit regarding primary containment pressure during normal operation to preserve the initial conditions assumed in the accident analysis for a DBA. This specification is applicable in Modes 1, 2, and 3. The limit on primary containment pressure was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.4 is not included 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. Thus, there will no longer be a need to maintain the limit on primary containment pressure.
TS 3.6.1.5, Primary Containment Air Temperature	This specification provides a limit regarding primary containment air temperature to preserve the initial conditions assumed in the accident analysis for a DBA. This specification is applicable in Modes 1, 2, and 3. The limit on primary containment air temperature was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.5 is not included 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. Thus, there will no longer be a need to maintain the limit on primary containment air temperature.
TS 3.6.1.6, Low-Low Set (LLS) Valves	This specification provides operability requirements for the LLS function of five S/RVs. The LLS function prevents excessive short duration S/RV cycles with valve actuation at the relief setpoint. The requirements of this LCO are applicable to the mechanical and electrical/pneumatic capability of the LLS valves to function for controlling the opening and closing of the S/RVs. This specification is applicable in Modes 1, 2, and 3. The LLS valves were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.6 is not included 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. Thus, there will no longer be a need for the LLS valves.
TS 3.6.1.7, Residual Heat Removal (RHR) Containment Spray System	This specification provides operability requirements for the RHR containment spray system. Its function was to mitigate the effects of drywell bypass leakage and low energy steam release into the primary containment airspace. This specification is applicable in Modes 1, 2, and 3. The RHR containment spray system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.7 is not included 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. Thus, there will no longer be a need for the RHR containment spray system.
TS 3.6.1.8 Deleted	Deleted
TS 3.6.1.9, Feedwater Leakage Control System (FWLCS)	This specification provides operability requirements for the FWLCS. The FWLCS supplements the isolation function of primary containment isolation valves (PCIVs) in the feedwater lines which also penetrate the secondary containment. These penetrations are sealed by water from the FWLCS to prevent fission products (post-LOCA containment atmosphere) from leaking past the isolation valves and bypassing the secondary containment after a DBA LOCA. This specification is applicable in Modes 1, 2, and 3. The FWLCS was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.1.9 is not included in the PDTS since CPS will be permanently shut down and defueled. The Part 50 license will prohibit operation of 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 the FWLCS.
TS 3.6.2.1, Suppression Pool Average Temperature	This specification provides a limit regarding suppression pool average temperature to assure that the primary containment conditions assumed for the safety analysis are met. This specification is applicable in Modes 1, 2, and 3. The limit on suppression pool average temperature was included in the ITS to satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.2.1 is not included 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. Thus, there will no longer be a need to maintain the limit on suppression pool average temperature.
TS 3.6.2.2, Suppression Pool Water Level	This specification provides a limit regarding suppression pool water level to preserve the initial conditions assumed in the accident analysis for a DBA. This specification is applicable in Modes 1, 2, and 3. The limit on suppression pool water level was included in the ITS to satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.2.2 is not included 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. Thus, there will no longer be a need to maintain the limit on suppression pool water level.
TS 3.6.2.3, Residual Heat Removal (RHR) Suppression Pool Cooling	This specification provides operability requirements for the RHR suppression pool cooling system. Its function was to remove heat from the suppression pool following a DBA. This specification is applicable in Modes 1, 2, and 3. The RHR suppression pool cooling system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.2.3 is not included 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. Thus, there will no longer be a need for the RHR suppression pool cooling system.
TS 3.6.2.4, Suppression Pool Makeup (SPMU) System	This specification provides operability requirements for the suppression pool makeup system. The function of the SPMU system was to transfer water from the upper containment pool to the suppression pool after a LOCA to maintain drywell horizontal vent coverage and an adequate suppression pool heat sink volume to ensure that the primary containment internal pressure and temperature stay within design limits. This specification is applicable in Modes 1, 2, and 3. The SPMU system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.2.4 is not included 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. Thus, there will no longer be a need for the suppression pool makeup system.
TS 3.6.3.1 Deleted	Deleted
TS 3.6.3.2, Primary Containment and Drywell Hydrogen Igniters	This specification provides operability requirements for the primary containment and drywell hydrogen igniters. The function of the hydrogen igniters was to reduce the hydrogen concentration in the primary containment following a degraded core accident by ensuring the combustion of hydrogen in a manner such that containment overpressure failure is prevented as a result of a postulated degraded core accident. This specification is applicable in Modes 1 and 2. The hydrogen igniters were included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.6.3.2 is not included 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. Thus, there will no longer be a need for the primary containment and drywell hydrogen igniters.
TS 3.6.3.3, Containment/ Drywell Hydrogen Mixing Systems	This specification provides the operability requirements for the Containment/Drywell Hydrogen Mixing System. The hydrogen mixing system ensures a uniformly mixed post-accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammability concentration. This specification is applicable in Mode 1 and 2. The containment/drywell

	hydrogen mixing system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.3.3 is not included 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. Thus, there will no longer be a need for the containment/drywell hydrogen mixing system.
TS 3.6.4.1, Secondary Containment	This specification provides the operability requirements for secondary containment. The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be operable, or that take place outside primary containment. This specification is applicable in Mode 1, 2, and 3, or during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel in the primary or secondary containment. The secondary containment was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.4.1 requires that secondary containment be operable for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) in the primary or secondary containment. TS 3.6.4.1 is not included in the PDTS since CPS will be permanently shut down and defueled. This specification will no longer be required after 24 hours of decay following shut down, because the nuclear fuel will no longer be considered to be "recently irradiated." In addition, the other condition requiring that secondary containment integrity be met (i.e., operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, the conditions requiring secondary containment integrity will no longer be applicable and secondary containment will not be required.
TS 3.6.4.2, Secondary Containment Isolation Dampers (SCIDs)	This specification provides the operability requirements for the SCIDs. The function of the SCIDs is to limit fission product release during and following postulated DBAs. This specification is applicable in Mode 1, 2, and 3, or during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel in the primary or secondary containment. The SCIDs were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.4.2 requires that secondary containment be operable for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core

	within the previous 24 hours). TS 3.6.4.2 is not included in the PDTS since CPS will be permanently shut down and defueled. This specification will no longer be required after 24 hours of decay following shut down, because the nuclear fuel will no longer be considered to be "recently irradiated." In addition, the other condition requiring operability of the SCIDs (i.e., operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, the conditions requiring the SCIDs to be operable will no longer be applicable and the SCIDs will not be required.
TS 3.6.4.3, Standby Gas Treatment (SGT) System	This specification provides the operability requirements for the SGT system. The function of the SGT system is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a DBA are filtered and adsorbed prior to exhausting to the environment. This specification is applicable in Mode 1, 2, and 3, or during operations with a potential for draining the reactor vessel, or during movement of recently irradiated fuel in the primary or secondary containment. The SGT system were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.4.3 requires that standby gas treatment be operable for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) in the primary or secondary containment. TS 3.6.4.3 is not included in the PDTS since CPS will be permanently shut down and defueled. This specification will no longer be required after 24 hours of decay following shut down, because the nuclear fuel will no longer be considered to be "recently irradiated." In addition, the other condition requiring operability of the SGT system (operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, the conditions requiring the SGT system to be operable will no longer be applicable and the SGT system will not be required.
TS 3.6.5.1, Drywell	This specification provides operability requirements for the drywell. Its function was to maintain a pressure boundary that channels steam from a LOCA to the suppression pool, where it is condensed and protect accessible areas of the containment from radiation originating in the reactor and reactor coolant system. This specification is applicable in Mode 1, 2, and 3. The drywell was included in the ITS to satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.5.1 is not included 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. Thus, there will no longer be a need for the drywell.
TS 3.6.5.2, Drywell Air Lock	This specification provides operability requirements for the drywell air lock. The airlock forms part of the drywell boundary and provides a means for personnel access during shutdown and the low power phase of startup. This specification is applicable in Mode 1, 2, and 3. The

	drywell air lock was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.5.2 is not included 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. Thus, there will no longer be a need for the drywell air lock.
TS 3.6.5.3, Drywell Isolation Valves	This specification provides operability requirements for the drywell isolation valves. Their function was to form a part of the drywell boundary and ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell. This specification is applicable in Mode 1, 2, and 3. The drywell isolation valves were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.6.5.3 is not included 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. Thus, there will no longer be a need for the drywell isolation valves.
TS 3.6.5.4, Drywell Pressure	This specification provides a limit regarding drywell-to-primary containment pressure which is an assumed initial condition in the analyses that determine the primary containment thermal hydraulic and dynamic loads during a postulated LOCA. This specification is applicable in Mode 1, 2, and 3. The limit on drywell pressure was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.6.5.4 is not included 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. Thus, there will no longer be a need to maintain the limit on drywell pressure.
TS 3.6.5.5, Drywell Air Temperature	This specification provides a limit regarding drywell air temperature to ensure that the peak drywell temperature during a design basis LOCA does not exceed the design temperature and ensures the ability of the drywell to perform its design function. This specification is applicable in Mode 1, 2, and 3. The limit on drywell air temperature was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.6.5.5 is not included 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. Thus, there will no longer be a need to maintain the limit on drywell air temperature.
TS 3.6.5.6, Drywell Post-LOCA Vacuum Relief System	This specification provides operability requirements for the drywell post-LOCA vacuum relief system. Its function was to support operation of the hydrogen mixing system and reduce suppression pool drag and impact loads in the event of a large break LOCA. This specification is applicable in Mode 1, 2, and 3. The drywell post-LOCA vacuum relief system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

	TS 3.6.5.6 is not included 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. Thus, there will no longer be a need for the drywell post-LOCA vacuum relief system.

TS SECTION 3.7, PLANT SYSTEMS (Current Title) **TS SECTION 3.7, FACILITY SYSTEMS (Proposed Title)** TS Section 3.7 contains LCOs and SRs that provide assurance of the safe operation of various plant systems. 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 systems. Following 60 days of decay post-shutdown, the dose consequences to occupants in the control room are acceptable without relying on the control room systems to remain functional during and following a FHA. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply (or are no longer needed) in a defueled condition are being proposed for deletion. TS 3.7.7, Spent Fuel Storage Pool Water Level, will be retained in the PDTS. In the Section Title, the reference to the term "plant" is replaced with the term "facility," because the term "plant" generally refers to the reactor, which can no longer be operated, whereas the term "facility" refers to the overall site.	
TS 3.7.1, Division 1 and 2 Shutdown Service Water (SX) Subsystems and Ultimate Heat Sink (UHS)	This specification provides the operability requirements for the Division 1 and 2 Shutdown Service Water (SX) Subsystems and the Ultimate Heat Sink (UHS). The function of the SX system is to provide cooling water for the removal of heat from unit auxiliaries, such as RHR system heat exchangers, standby DGs, and room coolers for ECCS equipment required for a safe reactor shut down following a DBA or transient. The UHS is a portion of Clinton Lake that provides sufficient water inventory for all SX system post-LOCA cooling requirements for a 30-day period with no external makeup water source available. This specification is applicable during Modes 1, 2 and 3. The SX system and ultimate heat sink were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.7.1 is not included 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. Thus, there will no longer be a need for the SX system.
TS 3.7.2, Division 3 Shutdown Service Water (SX) Subsystem	This specification provides the operability requirements for the Division 3 Shutdown Service Water (SX) Subsystem. The function of the Division 3 SX system is to provide cooling water for the removal of heat from components of the Division 3 HPCS system and ensure the HPCS system will operate as required. This specification is applicable during Modes 1, 2 and 3. The Division 3 SX system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.7.2 is not included 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. In addition, the specification requiring operability of the ECCS, including the HPCS system, is being proposed for

	deletion as described above in TS Section 3.5. Thus, there will no longer be a need for the Division 3 SX system.
TS 3.7.3, Control Room Ventilation System	This specification provides the operability requirements for the Control Room Ventilation System. The function of the control room ventilation system is to provide a protected environment from which occupants can control the plant following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The safety-related function of the Control Room Ventilation System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of recirculated air or outside supply air and a Control Room Envelope (CRE) boundary that limits the inleakage of unfiltered air. The Control Room Ventilation System is designed to maintain a habitable environment in the CRE for a 30 day continuous occupancy after a DBA, without exceeding 5 rem total effective dose equivalent (TEDE). This specification is applicable during Modes 1, 2 and 3. TS 3.7.3 also requires that the control room ventilation system be operable for situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during core alterations, or during movement of irradiated fuel assemblies in the primary or secondary containment. The control room ventilation system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.7.3 is not included 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 following shut down. 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 ventilation (CRV) systems. The results of the evaluation indicated a 60 day decay time to meet the regulatory acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183 without credit for CRV systems, intake radiation monitors, and operator action in selecting control room intake with the lowest concentration of airborne activity. In addition, following permanent removal of the fuel from the reactor vessel, operating modes and two of the three conditions requiring the operability of the CRV system (Core Alterations or operations with the potential to drain the reactor vessel) will not be applicable. The FHA analysis does not rely on the CRV system for accident mitigation (including any need for providing airborne radiological protection), the CRV system is therefore not required during movement of irradiated fuel assemblies for mitigation of a potential FHA. 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 design basis accident that is credible with the unit permanently defueled. As such, the requirement for the CRV system is being deleted because there are no design basis events that rely on the CRV system for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply. Therefore, after 60 days of decay following shutdown, the conditions requiring the CRV system to be operable will no longer be applicable.

TS 3.7.4, Control Room Air Conditioning (AC) System	This specification provides the operability requirements for the Control Room AC system. The function of the Control Room AC system is to provide temperature control for the control room and provide a controlled environment under both normal and accident conditions. This specification is applicable during Modes 1, 2 and 3. TS 3.7.4 also requires that the control room ventilation system be operable for situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during core alterations, or during movement of irradiated fuel assemblies in the primary or secondary containment. The Control Room AC system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.7.4 is not included 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 following shut down. Following permanent removal of the fuel from the reactor vessel, operating modes and two of the three conditions requiring the operability of the Control Room AC system (Core Alterations or operations with the potential to drain the reactor vessel) will not be applicable. The FHA analysis does not rely on the Control Room AC system for accident mitigation (including any need for maintaining control room temperature and habitability), the Control Room AC system is therefore not required during movement of irradiated fuel assemblies for mitigation of a potential FHA. 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 design basis accident that is credible with the unit permanently defueled. As such, the requirement for the Control Room AC system is being deleted because there are no design basis events that rely on the Control Room AC system for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply. Therefore, after 60 days of decay following shutdown, the conditions requiring the Control Room AC system to be operable will no longer be applicable.
TS 3.7.5, Main Condenser Offgas	This specification provides the operability requirements for the main condenser offgas system. The function of the main condenser offgas system is to reduce the gaseous radwaste emission. This specification is applicable during Modes 1 or in Mode 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation. The main condenser offgas system was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.7.5 is not included in the PDTS since CPS will be permanently shut down and defueled. The Part 50 license will prohibit operation of 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 the main condenser offgas system.
TS 3.7.6, Main Turbine Bypass System	This specification provides the operability requirements for the main turbine bypass system. The function of the main turbine bypass system is to control steam pressure when reactor steam generation exceeds turbine requirements during plant startup, sudden load reduction, and

	cool down. This specification is applicable when THERMAL POWER is ≥ 21.6% RTP. The main turbine bypass system was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.7.6 is not included 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. Thus, there will no longer be a need for the main turbine bypass system.
TS 3.7.7, Fuel Pool Water Level	This specification provides the minimum water level in the SFP and upper containment pool whenever movement of irradiated fuel assemblies occurs in the associated fuel storage pool since the potential for a release of fission products exists. TS 3.7.7 is being retained in the PDTS. The specification is administratively changed from affecting "plant" systems to "facility" systems to reflect a permanently defueled condition. The reference to the "upper containment fuel storage pool" is being deleted in the LCO statement. "Associated" fuel storage pool is being changed to "spent" fuel storage pool in the Applicability and the Required Action sections, since the upper containment pool will no longer be utilized once the reactor is permanently defueled and spent fuel is located in the SFP. The NOTE in Required Action A.1 (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS LCO 3.0.3 described in TS Section 3.0 LCO Applicability. The Frequency for SR 3.7.7.1 is being changed to 7 days to incorporate and conform with the deletion of the TS 5.5.16 Surveillance Frequency Control Program described in TS Section 5.0 Administrative Controls.

TS SECTION 3.8, ELECTRICAL POWER SYSTEMS

TS Section 3.8 contains LCOs and SRs that provide assurance of the integrity and safe operation of AC Sources, DC Sources, and Electrical Distribution Systems. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) will not apply (or are no longer needed) in a defueled condition. Therefore, TS Section 3.8 is proposed for deletion in its entirety.

TS 3.8.1, AC Sources - Operating	This specification provides the operability requirements for AC sources during specific operating Modes (i.e., Modes 1, 2, and 3). The function of the AC electrical power sources is to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to engineered safety feature systems (ESF) so that the fuel, RCS, and containment design limits are not exceeded. The AC sources were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.1 is not included 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. As a result, CPS will no longer be allowed to be in Modes 1, 2, and 3. Thus, there will no longer be a need for the AC sources during the specified Modes of operation.

TS 3.8.2, AC Sources - Shutdown	This specification provides the operability requirements for AC sources during specific shutdown Modes (i.e., Modes 4 and 5), or during the specified condition "During movement of irradiated fuel assemblies in the primary or secondary containment." The function of the AC electrical power sources is to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to engineered safeguards systems so that the fuel, RCS, and containment design limits are not exceeded. The AC sources were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.2 is not included 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. Thus, there will no longer be a need for this TS LCO during Modes 4 and 5. This TS LCO is a support TS LCO that supports TS LCO 3.8.10 Distribution Systems – Shutdown (AC). TS LCO 3.8.10 (AC) is a support TS LCO for TS LCOs 3.7.3 and 3.7.4. Once TS LCOs 3.7.3 and 3.7.4 specified requirement "During movement of irradiated fuel assemblies in the primary or secondary containment" is eliminated after 60 days of decay following shutdown, this support TS LCO can be eliminated.
TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air	This specification provides limits on the diesel fuel oil, lube oil, and starting air subsystems. These systems support the operation of the standby AC power sources in accordance with TS 3.8.1 and TS 3.8.2. This TS LCO is applicable whenever an associate EDG is required to be operable. The diesel fuel oil, lube oil, and starting air subsystems were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). As described above, TS 3.8.1 and TS 3.8.2 are not included in the PDTS. Thus, TS 3.8.3 is not included in the PDTS, because the TSs that it supports are no longer required after CPS is permanently shut down.
TS 3.8.4, DC Sources - Operating	This specification provides the operability requirements for DC sources during specific operating Modes (i.e., Modes 1, 2, and 3). The function of the DC electrical power sources is to provide the AC emergency power system with control power. It also provides both motive and control power to selected safety-related equipment. The DC sources were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.4 is not included 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. As a result, CPS will no longer be allowed to be in Modes 1, 2, and 3. Thus, there will no longer be a need for the DC sources during the specified Modes of operation.
TS 3.8.5, DC Sources – Shutdown	This specification provides the operability requirements for DC sources during specific shutdown Modes (i.e., Modes 4 and 5), or during the specified condition of "During the movement of irradiated fuel in the primary or secondary containment." The function of the DC electrical power sources is to provide the AC emergency power system with control power. It also provides both motive and control power to selected safety-related equipment. The DC sources were included in the ITS to

	satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.5 is not included 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. Thus, there will no longer be a need for this LCO during Modes 4 and 5. This TS LCO is a support TS LCO for TS LCO 3.8.10 Distribution Systems – Shutdown. The (DC) distribution system supports the (AC) distribution system. TS LCOs 3.8.10 supports TS LCOs for TSs 3.7.3 and 3.7.4. Once TS LCOs 3.7.3 and 3.7.4 requirement during the specified condition “During movement of irradiated fuel assemblies in the primary or secondary containment” is eliminated after 60 days of decay following shutdown, this support TS LCO can be eliminated.
TS 3.8.6, Battery Parameters	This specification provides limits on various battery parameters (i.e., battery float current, electrolyte temperature, level and float voltage). These support the batteries that are required to be operable in accordance with TS 3.8.4 and TS 3.8.5. The battery cell parameters were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). As described above, TS 3.8.4 and TS 3.8.5 are not included in the PDTS. Thus, TS 3.8.6 is not included in the PDTS, because the TS that it supports are no longer required after CPS is permanently shut down.
TS 3.8.7, Inverters - Operating	This specification provides the operability requirements for the Inverters during specific operating Modes (i.e., Modes 1, 2, and 3). The inverters are the preferred source of power for the uninterruptible AC buses and the Reactor Power System solenoid buses because of the stability and reliability they achieve. The inverters are powered from both AC and DC sources. The AC sources were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.7 is not included 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. As a result, CPS will no longer be allowed to be in Modes 1, 2, and 3. Thus, there will no longer be a need for the inverters during the specified Modes of operation.
TS 3.8.8, Inverters – Shutdown	This specification provides the operability requirements for the inverters during specific shutdown Modes (i.e., Modes 4 and 5), or during movement of irradiated fuel assemblies in primary or secondary containment. The inverters are the preferred source of power for the uninterruptible AC buses and the Reactor Power System solenoid buses because of the stability and reliability they achieve. The inverters are powered from both AC and DC sources. The AC sources were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.8 is not included 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. Thus, there will no longer be a need for the inverters during the specified Modes of operation (i.e. Modes 4 and 5). TS LCO 3.8.8 Inverters Shutdown is a support LCO that supports TS LCO 3.8.10 Distribution for the uninterruptable AC bus electrical distribution

	system. The uninterruptable AC distribution portion of TS LCO 3.8.10 is a support LCO that supports TS LCO's 3.3.1.1, 3.3.1.3, 3.3.2.1, 3.3.4.1, 3.3.4.2, 3.3.5.1, 3.3.5.2, 3.3.6.1, 3.3.6.2, 3.3.6.3, 3.3.6.4 and 3.3.6.5 which are being proposed for deletion, see individual discussions above. Therefore, since all of listed support TS LCO are being proposed for deletion this support TS LCO is being is being proposed for deletion.
TS 3.8.9, Distribution Systems - Operating	This specification provides the operability requirements for AC and DC distribution systems during specific operating Modes (i.e., Modes 1, 2, and 3). The function of the AC and DC and uninterruptible AC bus electrical power distribution systems provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature systems so that the fuel, RCS, and containment design limits are not exceeded. The AC and DC distribution systems were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.9 is not included 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. As a result, CPS will no longer be allowed to be in Modes 1, 2, and 3. Thus, there will no longer be a need for AC and DC distribution systems during the specified Modes of operation. .
TS 3.8.10, Distribution Systems – Shutdown	This specification provides the operability requirements for AC and DC distribution systems during specific shutdown Modes (i.e., Modes 4 and 5), or during the specified condition of "During the movement of irradiated fuel assemblies in primary or secondary containment." The function of the AC and DC and uninterruptible AC bus electrical power distribution systems provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature systems so that the fuel, RCS, and containment design limits are not exceeded. The AC and DC distribution systems were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.8.10 is not included 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. Thus, there will no longer be a need for the distribution systems TS LCO during the specified Modes 4 or 5. This TS LCO is a support TS LCO that supports TS LCOs 3.7.3 and 3.7.4. Once TS LCOs 3.7.3 and 3.7.4 specified condition of "During movement of irradiated fuel assemblies in the primary or secondary containment" is eliminated after 60 days of decay following shutdown, this support LCO can be eliminated.
TS 3.8.11, Static VAR Compensator (SVC) Protection Systems.	This specification provides the operability requirements for the SVC protection systems. Each of the auxiliary power transformers is provided with a permanently installed SVC which can be connected to the secondary side of the transformer to provide steady state, dynamic, and transient voltage support. The internal control system for each SVC includes control and protective functions. Backup protection is provided by a fully redundant and independent protection system. The protection systems are intended to minimize the potential for SVC failures to damage or degrade required ESF equipment. The SVC Protection

	systems were included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). According to TS 3.8.11 the SVC protection system must be operable whenever its associated SVC is in operation. The SVC operation is associated with the operability of the offsite sources as governed by TS 3.8.1 and 3.8.2. As described above, TS 3.8.1 and TS 3.8.2 are not included in the PDTS. Therefore, TS 3.8.11 is not included in the PDTS, because the TSs that it supports are no longer required after CPS is permanently shut down. Additionally, all of the accident sequences that previously dominated risk at CPS do not use AC power and will no longer be applicable once the reactor is in the permanently shut down and defueled condition.

TS SECTION 3.9, REFUELING OPERATIONS	
TS Section 3.9 contains LCOs and SRs that provide operability requirements for systems and components and limits to: 1) prevent reactivity excursions; 2) limit offsite doses from an accident; and 3) remove decay heat from the RCS. These specifications are applicable during refueling operations. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) in Section 3.9 will no longer apply in a defueled condition. Therefore, TS Section 3.9 is proposed for deletion in its entirety.	
TS 3.9.1, 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 was to restrict the operation of the refueling equipment or the withdrawal of control rods to reinforce plant procedures that prevent the reactor from achieving criticality during refueling. The refueling equipment interlocks were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.1 is not included 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. Once the CPS reactor has been permanently defueled, refueling equipment interlocks will no longer be necessary since the interlocks are only required to be operable during in-vessel fuel movement with refueling equipment associated with the interlocks. Thus, there will no longer be a need for this specification.
TS 3.9.2, Refuel Position One-Rod-Out Interlock	This specification provides the operability requirements for the refuel position one-rod-out interlock. This specification is applicable in Mode 5 with the reactor mode switch in the refuel position and any control rod withdrawn. Its function was to restrict the movement of control rods to reinforce plant procedures that prevent the reactor from becoming critical during refueling operations. The refuel position one-rod-out interlock was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.2 is not included 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. Once the CPS reactor has been permanently defueled, the need to prevent an inadvertent criticality during refueling activities will no longer be a concern. Thus, there will no longer be a

	need for this specification.
TS 3.9.3, Control Rod Position	This specification requires all control rods to be fully inserted when loading fuel assemblies into the core to minimize the probability of an inadvertent criticality during refueling. This specification is applicable when loading fuel assemblies into the core. The control rod position limit was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.3 is not included 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. Once the CPS reactor has been permanently defueled, control rod position will no longer be necessary since the control rod position limitations are only required during loading of fuel into the core. Thus, there will no longer be a need for this specification.
TS 3.9.4, Control Rod Position Indication	This specification provides operability requirements for the control rod full-in position indication channel for each control rod to provide the required inputs to the refueling interlocks. This specification is applicable during Mode 5. Control rod position indication was included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.4 is not included 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. Once the CPS reactor has been permanently defueled, applicability of this requirement will never be entered again. Control rod position indication in relation to the fuel will no longer be necessary.
TS 3.9.5, Control Rod OPERABILITY - Refueling	This specification provides operability requirements for withdrawn control rods during refueling to ensure the control rods will insert and provide the required negative reactivity to maintain the reactor subcritical. This specification is applicable during Mode 5. These control rod requirements were included in the ITS to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). TS 3.9.5 is not included 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. Once the CPS reactor has been permanently defueled, applicability of this requirement will never be entered again. Thus, there will no longer be a need for this specification.
TS 3.9.6, Reactor Pressure Vessel (RPV) Water Level – Irradiated Fuel	This specification provides a limit regarding the RPV water level during movement of irradiated fuel assemblies within the RPV. This specification is applicable during movement of irradiated fuel assemblies within the RPV. The RPV water level limit was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.9.6 is not included 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. Once the CPS reactor is permanently defueled, there will no longer be any movement of irradiated fuel assemblies in the

	RPV. Since the water level limits specified in this TS are only required during refueling activities, there will no longer be a need for this specification.
TS 3.9.7, Reactor Pressure Vessel (RPV) Water Level – New Fuel or Control Rods	This specification provides a limit regarding the RPV water level during movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the RPV are irradiated. The RPV water level limit was included in the ITS to satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). TS 3.9.7 is not included 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. Once the CPS reactor is permanently defueled, there will no longer be any new or irradiated fuel within the RPV. Since the water level limits specified in this TS are only required during refueling activities, there will no longer be a need for this specification.
TS 3.9.8, Residual Heat Removal (RHR) – High Water Level	This specification provides operability requirements for the RHR shutdown cooling subsystem during Mode 5 with irradiated fuel in the RPV and the water level ≥ 22 ft. 8 in. above the RPV flange. The purpose of the RHR System in MODE 5 is to remove decay heat and sensible heat from the reactor coolant. These RHR shut down cooling requirements were included in the ITS to satisfy Criterion 4 of 10 CFF 50.36(c)(2)(ii). TS 3.9.8 is not included 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. Once the CPS reactor is permanently defueled, there will no longer be any refueling activities taking place. Following permanent defueling there will be no irradiated fuel in the RPV or need for RHR shutdown cooling. Thus, there will no longer be a need for this specification.
TS 3.9.9, Residual Heat Removal (RHR) – Low Water Level	This specification provides operability requirements for the RHR shut down cooling subsystem during Mode 5 with irradiated fuel in the RPV and the water level < 22 ft. 8 in. above the RPV flange. The purpose of the RHR System in MODE 5 is to remove decay heat and sensible heat from the reactor coolant. These RHR shut down cooling requirements were included in the ITS to satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). TS 3.9.9 is not included in the PDTS 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. Once the CPS reactor is permanently defueled, there will no longer be any refueling activities taking place. Following permanent defueling there will be no irradiated fuel in the RPV or need for RHR shutdown cooling. Thus, there will no longer be a need for this specification.

TS SECTION 3.10, SPECIAL OPERATIONS
TS Section 3.10 contains Special Operations LCOs and SRs that provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. Because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated

SRs) will no longer apply in a defueled condition. Therefore, TS Section 3.10 is proposed for deletion in its entirety.	
TS 3.10.1, Inservice Leak and Hydrostatic Testing Operation	This specification provides the requirements to allow certain reactor coolant pressure tests to be performed in Mode 4 when the metallurgical characteristics of the RPV require the pressure testing at temperatures > 200°F (normally corresponding to Mode 3). As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.1 is not included in the PDTS 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. Once the CPS reactor is permanently defueled, the reactor vessel will no longer be within the applicability condition of Modes 3 or 4, since modes will no longer be applicable. Thus, there will no longer be a need for this specification.
TS 3.10.2, Reactor Mode Switch Interlock Testing	This specification provides the requirements to permit operation of the reactor mode switch from one position to another to confirm certain aspects of associated interlocks during periodic tests and calibrations in Modes 3, 4, and 5. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.2 is not included 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. This specification allows reactor mode switch interlock testing during Modes 3, 4 and 5 that cannot conveniently be performed without this allowance. As described above, once the CPS reactor is permanently defueled, the operating modes will no longer be applicable. The need to test the reactor mode switch interlocks under any conditions will also no longer be necessary. Thus, there will no longer be a need for this specification.
TS 3.10.3, Single Control Rod Withdrawal - Hot Shut down	This specification provides the requirements and additional controls to permit the withdrawal of a single control rod for testing while in hot shut down (Mode 3 with the reactor mode switch in the Refuel position), by imposing certain restrictions. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.3 is not included 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. Since the control rod withdrawal in Mode 3 requires controls consistent with those required in refueling, once the CPS reactor is permanently defueled, there will no longer be any

	refueling activities taking place and therefore no restrictions on the withdrawal of control rods. As described above, once the CPS reactor is permanently defueled, the operating modes will no longer be applicable. The need to implement controls to withdraw a single control rod in Mode 3 will no longer be required. Thus, there will no longer be a need for this specification.
TS 3.10.4, Single Control Rod Withdrawal - Cold Shut down	This specification provides the requirements to permit the withdrawal of a single control rod for testing or maintenance, while in cold shut down (Mode 4 with the reactor mode switch in the Refuel position), by imposing certain restrictions. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.4 is not included 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. Since the control rod withdrawal in Mode 4 requires controls consistent with those required in refueling, once the CPS reactor is permanently defueled, there will no longer be any refueling activities taking place and therefore no restrictions on the withdrawal of control rods. As described above, once the CPS reactor is permanently defueled, the operating modes will no longer be applicable. The need to implement controls to withdraw a single control rod in Mode 4 will no longer be required. Thus, there will no longer be a need for this specification.
TS 3.10.5, Single Control Rod Drive (CRD) Removal - Refueling	This specification provides the requirements to permit the removal of a single CRD during refueling operations by imposing certain administrative controls. Applicable condition Mode 5 with TS 3.9.5 not met. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.5 is not included 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. Once the CPS reactor is permanently defueled, there will no longer be any mode in effect. Since the requirements specified in this TS are only required during refueling, there will no longer be a need for this specification.
TS 3.10.6, Multiple Control Rod Withdrawal - Refueling	This specification provides the requirements to permit multiple control rod withdrawal during refueling by imposing certain administrative controls. Applicable condition Mode 5 with TS 3.9.3, 3.9.4, and 3.9.5 not met. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.6 is not included 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. Once the CPS reactor is permanently defueled, there will no longer be any mode in effect. Since the requirements specified in this TS are only required during refueling, there will no longer be a need for this specification.
TS 3.10.7, Control Rod Testing - Operating	This specification provides the requirements to permit control rod testing, while in Modes 1 and 2, by imposing certain administrative controls. This Special Operations LCO provides the necessary exceptions to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.7 is not included in the PDTS 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. Once CPS is shutdown and permanently defueled there will no longer be a need to perform control rod testing. In addition, CPS will no longer be permitted to enter Modes 1 or 2 in accordance with 10 CFR 50.82 (a)(2). Thus, there will no longer be a need for this specification.
TS 3.10.8, SHUT DOWN MARGIN (SDM) Test-Refueling	This specification provides the requirements to permit SDM testing to be performed for those plant configurations in which the RPV head is either not in place or the head bolts are not fully tensioned (Mode 5). LCO 3.1.1, "Shutdown Margin (SDM)," requires that adequate SDM be demonstrated following fuel movements or control rod replacement within the RPV. The demonstration must be performed prior to or within 4 hours after criticality is reached. This SDM test may be performed prior to or during the first startup following refueling. Performing the SDM test prior to startup requires the test to be performed while in MODE 5 with the vessel head bolts less than fully tensioned (and possibly with the vessel head removed). As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.8 is not included 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. As described above, Exelon is proposing to delete the definition of SDM in TS 1.1 and delete TS 3.1.1 since it will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted. Once CPS is shutdown and permanently defueled there will no longer be a need to perform SDM testing since refueling activities will no longer take place. Since the requirements specified in this TS are only required during refueling, there will no longer be a need for this specification.

TS 3.10.9, Training Startups	This specification provides the requirements to permit training startups to be performed while in Mode 2 to provide plant startup experience for reactor operators. This training involves withdrawal of control rods to achieve criticality and then further withdrawal of control rods, as would be experienced during an actual plant startup. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.9 is not included 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. Based on the fact that CPS will no longer be operated, training of operators in plant startup will no longer be required.
TS 3.10.10, Single Control Rod Withdrawal - Refueling	This specification provides the requirements to permit the withdrawal of a single control rod for testing in Mode 5 without imposing the requirements for establishing the secondary containment and main control room boundaries as normally required during Core Alterations. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3.10.10 is not included 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. Once the CPS reactor is permanently defueled, there will no longer be any Core Alterations taking place in the reactor vessel; therefore, restrictions or consequences for withdrawal of control rods will no longer be necessary. As described above, once the CPS reactor is permanently defueled, the operating modes will no longer be applicable. The need to implement controls to withdraw a single control rod in Mode 5 will no longer be required. Thus, there will no longer be a need for this specification.

TS SECTION 4.0, DESIGN FEATURES	
The existing TS Section 4.0, “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 CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the design features that do not apply in a defueled condition are being proposed for deletion.	
TS 4.2, Reactor Core	TS 4.2 is being proposed for elimination. The specification has two sections: TS 4.2.1, Fuel Assemblies TS 4.2.1 provides a description and requirements regarding the reactor core fuel assemblies. This TS does not apply in a defueled condition and will not be retained in the PDTS, because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel

	once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Thus, without fuel in the reactor vessel there is no need for the specification. TS 4.2.2, Control Rod Assemblies TS 4.2.2 provides a description and requirements regarding the control rod assemblies in a reactor core. This TS does not apply in a defueled condition and will not be retained in the PDTS, because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Thus, without fuel in the reactor vessel there is no need for the specification.
TS 4.3, Fuel Storage	TS 4.3 provides a description and requirements regarding prevention of criticality of spent fuel, prevention of spent fuel pool drainage, and spent fuel capacity limitations. TS 4.3.1, Criticality TS 4.3.1.1.a will be retained. The specification requires that the K_{eff} of the fuel in the spent fuel pool be less than or equal to 0.95 with unborated water. The basis for this requirement is described in the USAR Section 9.1.2. 4.3.1.1.b is proposed for elimination. This specification describes the spacing of fuel assemblies stored in the upper containment pool. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, all spent fuel will be stored in the SFP. The containment upper pool will no longer be used. TS 4.3.1.1.c will be reformatted to be TS 4.3.1.1.b. TS 4.3.1.2 provides a description and requirements regarding the design of the new fuel storage racks. This description is being proposed for deletion since new fuel is no longer stored onsite and License Condition 2.B.(3) is being revised to remove the allowance to receive new fuel. The design feature associated with the new fuel storage racks is no longer applicable and may be deleted. TS 4.3.3, Capacity TS 4.3.3.2 provides a limit on the number of fuel assemblies that can be stored in the upper containment pool. Once the reactor is certified permanently defuel pursuant to 10 CFR 50.82(a)(2), all spent fuel will be stored in the SFP. The containment upper pool will no longer be used; therefore, TS 4.3.3.2 can be deleted.

TS SECTION 5.0, ADMINISTRATIVE CONTROLS

The existing TS Section 5.0, 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. Previously, Exelon proposed changes to the administrative controls in TS Sections 5.1, 5.2, 5.3, and 5.4 to reflect the permanently shut down and defueled state in a letter dated July 28, 2016 (Reference 2) as supplemented in Reference 3.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, several of the TS Section 5.0 Specifications are no longer applicable. Therefore, the administrative controls that

do not apply in a defueled condition are being proposed for deletion.	
TS Section 5.1 - Responsibility	Exelon proposed changes to the administrative controls in TS Section 5.1 to reflect the permanently shut down and defueled state in a letter dated July 28, 2016 (Reference 2).
TS Section 5.2 - Organization	Exelon proposed changes to the administrative controls in TS Section 5.2 to reflect the permanently shut down and defueled state in a letter dated July 28, 2016 (Reference 2) as supplemented in Reference 3.
TS Section 5.3 - Unit Staff Qualifications	Exelon proposed changes to the administrative controls in TS Section 5.3 to reflect the permanently shut down and defueled state in a letter dated July 28, 2016 (Reference 2). TS 5.3.1 is being revised to add clarification for staff qualification. With the transition to decommissioning certain personnel and titles for positions not stated in ANSI/ANS 3.1-1978 will be used. The comparable positions and exceptions will be specified in the Quality Assurance Program Manual (QAPM).
TS 5.4 Procedures	TS 5.4.1.a - Exelon proposed changes to this TS Section to reflect the permanently shut down and defueled state in a letter dated July 28, 2016 (Reference 2). TS 5.4.1.b - This TS Section requires emergency operating procedures be established, implemented, and maintained that implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 (ADAMS Accession No. ML051400209), and NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983 (ADAMS Accession No. ML 102560009), as stated in Generic Letter 82-33, "Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability," dated December 17, 1982 (ADAMS Accession No. ML031080548), incorporated into one document all Three Mile Island (TMI) related items approved for implementation by the NRC at that time. This included the use of human factored, function oriented, emergency operating procedures to improve human reliability and the ability to mitigate the consequences of a broad range of initiating events for operating reactors, and subsequent multiple failures or operator errors, without the need to diagnose specific events. TS 5.4.1.b is proposed for deletion because the emergency operating procedures discussed therein only pertain to accidents and events resulting from reactor operation. The referenced procedures will no longer be required for a permanently shutdown and defueled reactor. TS 5.4.1.c and 5.4.1.d – These TS Sections will be retained without change.
5.5.2, Primary Coolant Sources 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 to levels as low as practicable. It addresses portions of the LPCS, HPCS, RHR, RCIC, Suppression Pool Makeup Water, Combustible Gas Control,

	Containment Monitoring, and Post-Accident Sampling Systems. As previously discussed, the TS requirements for these systems were proposed for deletion. Once the plant is permanently shut down and defueled, there will no longer be any transient or accident conditions associated with primary coolant sources. Thus, TS 5.5.2 will not be retained in the PDTS.
5.5.4, Radioactive Effluent Controls Program	This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program is contained in the ODCM. The term "unit" is being changed to "facility" to be more appropriate with a defueled condition, which is consistent with other changes in this LAR.
5.5.5, Component Cyclic or Transient Limit	This program provides controls to track the cyclic and transient occurrences to ensure that the reactor vessel is maintained within the design limits. The program will not be retained in the PDTS, because the CPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The reactor vessel will no longer be subjected to cycles or transients after permanent shutdown.
5.5.6, Inservice Testing Program	TS 5.5.6 will not be retained in the PDTS. This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. In the permanently shut down and defueled condition, there are no longer any ASME Code class pumps and valves that remain in operation and are relied upon to mitigate a DBA. As such, the inservice testing program will no longer be relevant to CPS. Exelon has proposed changes to delete this program in a letter dated October 6, 2016 (Reference 14) to incorporate TSTF-545 which is currently under NRC review.
5.5.7, Ventilation Filter Testing Program (VFTP)	This program was established to implement the required testing of the filter ventilation systems for the SGTS and Control Room ventilation systems. This program will not be retained in the PDTS, because the VFTP is no longer required in a permanently shut down and defueled condition. As previously discussed, TSs 3.6.4.3 and 3.7.3 that provided the operability requirements for the SGT System and the CRV System were proposed to be eliminated.
5.5.8, Explosive Gas and Storage Tank Radioactivity Monitoring Program	This program provides controls for potentially explosive gas mixtures contained in the Radioactive Waste Disposal System, and the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system. The requirements regarding the Storage Tank Radioactivity Monitoring Program will be retained. The requirements regarding explosive gas mixtures contained in the main condenser offgas treatment system are not applicable during the permanently shut down and defueled condition. There will no longer be any source of explosive gas generated from reactor operation.
5.5.9, Diesel Fuel Oil Testing Program	The Diesel Fuel Oil Testing Program was established to implement the required testing of both new and stored fuel oil for the emergency diesel generators. This program is proposed for elimination from the PDTS.

	As described above, TSs 3.8.1, 3.8.2 and 3.8.3 are not included in the PDTS. Therefore, TS 5.5.9 is not required to be included in the PDTS.
5.5.10, Safety Function Determination Program (SFDP)	This program was established to ensure loss of safety function is detected and appropriate actions taken. The SFDP is proposed for elimination since the LCOs remaining in the PDTS do not rely on the operability of any active equipment or systems to satisfy the LCO. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, there is no longer a need for redundant systems. Therefore, the requirements of the SFDP, which directs cross train checks of multiple and redundant safety systems, no longer apply. Additionally, the SFDP is invoked in LCO 3.0.6, which is being deleted in its entirety as previously discussed. Thus, the SFDP is not needed in a permanently shut down and defueled condition.
5.5.12, Ultimate Heat Sink (UHS) Erosion, Sediment Monitoring, and Dredging Program	This program was established to provide maintenance on the UHS in the event inspections of the UHS dam, its abutments, or the UHS shoreline indicate erosion or local instability. This program supports TS 3.7.1 which is being eliminated as described above. Therefore, TS 5.5.12 is not required to be included in the PDTS.
5.5.13, Primary Containment Leakage Rate Testing Program	This program was established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by exemptions. This program will not be retained in the PDTS, because the Primary Containment Leakage Rate Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirements in TSs 3.6.1.1, 3.6.1.2, and 3.6.1.3, for primary containment systems are being deleted as described in preceding sections. Therefore, the need for leakage rate testing of primary containment is no longer applicable.
5.5.14, Battery Monitoring and Maintenance Program	This program was established to provide for battery restoration and maintenance. "Battery Monitoring and Maintenance Program," is being deleted because the Battery Monitoring and Maintenance Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirement for station batteries is being deleted as described in the discussions for TS 3.8.4, 3.8.5, and 3.8.6. The revised FHA in the SFP analysis applicable to the permanently defueled condition does not rely on batteries for accident mitigation.
5.5.15, Control Room Envelope Habitability Program	This program was established and implemented to ensure that the CRE habitability was maintained such that, with an operable CRV system, the occupants of the CRE can control the reactor safely under normal and emergency conditions and maintain it in a safe condition following a radiological event, hazardous chemical release or a smoke challenge. 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 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 3.7.3 and 3.7.4 will not be retained in the PDTS. Thus, TS 5.5.15 will not be retained in the PDTS.
5.5.16, 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) will not be retained in the PDTS. The remaining TS LCO proposed for this PDTS contains one surveillance requirement (SR). The proposed SR has been changed to reflect the frequency from the SFCP; therefore, no further need to maintain this program exists and it can be eliminated.
5.6.2, Annual Radiological Environmental Operating Report	This reporting requirement is being retained in the PDTS with essentially no change. The reference to the term "unit" is replaced with the term "facility," because the term "unit" generally refers to the reactor, which can no longer be operated in accordance with 10 CFR 50.82(a)(2), whereas the term "facility" refers to the overall site.
5.6.3, Radioactive Effluent Release Report	This reporting requirement is being retained in the PDTS with essentially no change. The reference to the term "unit" is replaced with the term "facility," because the term "unit" generally refers to the reactor, which can no longer be operated in accordance with 10 CFR 50.82(a)(2), whereas the term "facility" refers to the overall site.
5.6.5, CORE OPERATING LIMITS REPORT (COLR)	According to TS 5.6.5, the COLR is established prior to each reload cycle or prior to any remaining portion of a reload cycle to document the specific limits associated with operating the reactor core and to ensure that the applicable limits of the safety analysis are met. This reporting requirement will not be retained in the PDTS, because the Part 50 license will prohibit operation of the reactor or placement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Thus, the COLR does not apply in the permanently shut down and defueled condition.

The proposed changes are shown on the marked-up CPS 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 June 20, 2016 (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 CPS by June 1, 2017.

10 CFR 50.36 *"Technical specifications."*

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission 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 five 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. However, the rule does not specify the particular requirements to be included in a plant's TS.

The final Commission Policy Statement established four criteria to define the scope of equipment and parameters to be included in the improved Standard Technical Specifications. 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 Technical Specifications 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 Technical Specification 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 reactor coolant system at the CPS 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 CPS in the permanently defueled condition, the FHA, 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 SSCs that are 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 CPS, there are still DBAs that will 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 scope of DBAs that will be applicable to CPS is discussed 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 SSCs that 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 CPS 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 CPS 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 June 20, 2016 (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 CPS by June 1, 2017. Upon submittal of the final certification that fuel has been permanently removed from the CPS reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii), CPS will no longer be licensed to operate. The proposed amendment deletes the portions of the previous CPS TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shut down condition.

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 Commission 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 Commission notifies Exelon in writing that the license is terminated.

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 50.62, *"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 50.2 *"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 4), the NRC approval of a Certified Fuel Handler training program for CPS.

3.2 <u>Precedent</u>

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 7); (2) Kewaunee Power Station, for which an amendment was issued on February 13, 2015 (Reference 8); (3) San Onofre Nuclear Generating Station, Units 2 and 3, for which an amendment was issued on July 17, 2015 (Reference 9); (4) Zion Nuclear Power Station, for which an amendment was issued on December 30, 1999 (Reference 10); and (5) Crystal River Nuclear Plant, Unit 3, for

which an amendment was issued on September 4, 2015 (Reference 11).

3.3 No Significant Hazards Consideration

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 Facility Operating License (FOL) and Appendix A, Technical Specifications (TS), of FOL No. NPF-62 for Clinton Power Station (CPS).

The proposed amendment would revise the FOL 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 FOL 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 CPS reactor in accordance with 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 CPS no longer will authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2). The proposed changes to the FOL 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 CPS 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 CPS 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 CPS FOL 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 CPS Updated Safety Analysis Report (USAR) described the design basis accident (DBA) and transient scenarios applicable to CPS during power operations. Once the reactor is in a permanently defueled condition, the spent fuel pool 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 CPS in accordance with 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 in accordance with 10 CFR 50.82(a)(2), the majority of the accident scenarios previously postulated in the USAR will no longer be possible and will be removed from the USAR 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 CPS has no impact on the remaining applicable DBAs, the Fuel Handling Accident (FHA) or the Postulated Radioactive Releases Due to Liquid Radwaste Tank Failures. 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 CPS 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 CPS.

After CPS 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 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 spent fuel pool (SFP) water level and 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 CPS will no longer be authorized to operate the reactor.

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

The TS regarding SFP water level and fuel storage required 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 CPS 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 CPS 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 DBA.

The proposed changes are limited to those portions of the FOL 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 CPS FOL and TS are not credited in the existing accident analysis for the remaining applicable postulated accident; 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 CPS 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 Conclusion

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 Commission’s regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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 CPS. 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 Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Certification of Permanent Cessation of Power Operations," dated June 20, 2016 (ML16172A137)
2. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition," dated July 28, 2016 (ML16210A300)
3. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition – Supplement 1," dated November 4, 2016 (ML16309A013)
4. Letter from U.S. NRC 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 Patrick R. Simpson (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Request for NRC to Suspend Review of License Amendment Request for Clinton Power Station, Unit 1," dated August 11, 2016 (ML16224A262)
6. NUREG-1434, "Standard Technical Specifications - General Electric Plants (BWR/6)," Revision 4 (ML12104A195)
7. Letter from U.S. NRC 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)
8. Letter from U.S. NRC 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)
9. Letter from U.S. NRC 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)
10. Zion Nuclear Station, Units 1 and 2, License Amendments 180 and 167 to Facility Operating License Nos. DPR-39 and DPR-48, respectively, dated December 30, 1999 (ADAMS Accession Nos. ML003672704 and ML003672696)
11. Letter from U.S. NRC 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)
12. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for Amendment to Technical Specifications that Revise Plant System Requirements

During Fuel Handling Based on Alternate Source Term," dated July 5, 2001 (ML011980184)

13. Letter from U.S. NRC to Mr. Oliver D. Kingsley (Exelon Generation Company, LLC), "Clinton Power Station, Unit 1 – Issuance of Amendment (TAC No. MB2572)," dated April 3, 2002 (ML020930117)
14. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Supplement Application to Revise Technical Specifications to Adopt TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing"," dated October 6, 2016 (ML16280A402)
15. Letter from Christopher J. Wamser (Entergy Nuclear Operations, Inc), to U.S Nuclear Regulatory Commission, "Technical Specifications Proposed Change No. 309, Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition - Supplement 5 (TAC No. MF3714)," dated May 4, 2015 (ML15127A171)
16. 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)
17. Letter from K. N. Jabbour (U.S. NRC) to C. M. Crane (AmerGen Energy Company, LLC), "Clinton Power Station, Unit 1 – Issuance of an Amendment – Re: Onsite Spent Fuel Storage Expansion (TAC NO. MC4202)," dated October 31, 2005 (ML053070593)
18. EPRI Report 1025204, "Strategy for Managing the Long Term Use of Boral® in Spent Fuel Storage Pools," July 2012

Attachment 2

Proposed Technical Specifications (Marked-Up Pages)

Clinton Power Station
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Changes to the Facility Operating License and associated CPS Technical Specification (TS).
(74 pages)

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TS 3.7.7	Fuel Pool Water Level
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TS 5.1	Responsibility
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TS 5.4	Procedures
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Attachment 3

Proposed Technical Specifications Bases (Marked-Up Pages)

Clinton Power Station
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Changes to the CPS Technical Specification (TS) Bases pages
(TS Bases not listed are deleted in their entirety).
(11 Pages)

TS B 3.0	"Limiting Condition for Operation (LCO) Applicability"/ "Surveillance Requirement (SR) Applicability"
TS B 3.7.7	"Fuel Pool Water Level"

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

EXELON GENERATION COMPANY, LLC

DOCKET NO 50-461

CLINTON POWER STATION, UNIT NO. 1

FACILITY OPERATING LICENSE

License No. NPF-62

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

A. The application for license 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 regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

Deleted

B. ~~Construction of the Clinton Power Station, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-137 and the application, as amended, the provisions of the Act and the regulations of the Commission;~~

is prohibited from operating the reactor

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

D. 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 regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D below);

E. Exelon Generation Company, LLC (Exelon Generation Company) is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-62, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied;

I. The receipt, possession, and use of source, byproduct, and special nuclear material 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 receipt, production, possession, transfer, and use of Cobalt-60 as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30.

2. Based on the foregoing findings regarding this facility, and pursuant to approval by the Nuclear Regulatory Commission at a meeting on April 10, 1987, Facility Operating License No. NPF-62, which supersedes the license for fuel loading and low power testing, License No. NPF-55, issued on September 29, 1986, is hereby issued to Exelon Generation Company to read as follows:

A. This license applies to the Clinton Power Station, Unit No. 1, a boiling water nuclear reactor and associated equipment (the facility), owned by Exelon Generation Company. The facility is located in Harp Township, DeWitt County, approximately six miles east of the city of Clinton in east-central Illinois and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report-Operating License Stage, as supplemented and amended.

B. Subject to the condition and requirements incorporated herein, the Commission hereby licenses:

(1) Exelon Generation Company, pursuant to section 103 of the Act and 10 CFR Part 50, to possess~~,~~ and use ~~and operate~~ the facility at the designated location in Harp Township, DeWitt County, Illinois, in accordance with the procedures and limitations set forth in this license;

(2) Deleted

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess ~~and to 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 Final Safety Analysis Report, as supplemented and amended;

that were used   that were used

(4) Exelon Generation Company, pursuant to the Act and to 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material 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;

calibration   are used in

(5) Exelon Generation Company, 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 material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

that were

(6) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~ produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation; and

(7) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

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

(1) Maximum Power Level

Deleted

~~Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3473 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.~~

(2) Technical Specifications and Environmental Protection Plan

[insert Amendment #]

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ~~208~~, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan. |

(3) Antitrust Conditions

Deleted

(4) Control System Failures (Section 7.7.3.1, SER and SSER 6)*

Deleted

Deleted

(5) New Fuel Storage (Section 9.1.1, SER, SSER 6 and SSER 7)

~~Exelon Generation Company shall store new fuel assemblies in accordance with the requirements specified in Attachment 2. Attachment 2 is hereby incorporated into this license.~~

(6) Plant Operation Experience (Section 13.1.2.1, SSER 5)

Deleted

(7) Emergency Planning (Section 13.3, SSER 6)

Deleted

(8) Post-Fuel Loading Initial Test Program (Section 14, SER, SSER 5 and SSER 6)

Deleted

(9) Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737, Section 7.5.3.1, SSER 5 and SSER 8, and Section 18, SER, SSER 5 and Safety Evaluation Dated April 17, 1987)

a. Deleted

b. Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(22) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the follow 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

Deleted

(23) ~~Upon implementation of Amendment No. 178 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.3.5, in accordance with TS 5.5.15.c.(i), the assessment of CRE habitability as required by Specification 5.5.15.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.15.d, shall be considered met. Following implementation:~~

~~(a) The first performance of SR 3.7.3.5, in accordance with Specification 5.5.15.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from November 16, 2004, the date of the most recent successful tracer gas test, as stated in the February 8, 2005 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~

~~(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.15.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from November 16, 2004, the date of the most recent successful tracer gas test, as stated in the February 8, 2005 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~

(c) ~~The first performance of the periodic measurement of CRE pressure, Specification 5.5.15.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test, or within 6 months if not performed previously.~~

(24) At the time of the closing of the transfer of CPS 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 Clinton 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 CPS. Furthermore, funds dedicated for CPS prior to closing shall remain dedicated to CPS following the closing. The name of AmerGen Clinton NQF, LLC shall be changed to Exelon Generation Clinton NQF, LLC at the time of the closing.

(25) Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

D. The facility requires ~~exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a)~~ an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storage area~~; (b) an exemption from the requirement of 10 CFR Part 50, Appendix J – Option B, paragraph III.B, exempting the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for local leak rate tests (Section 6.2.6 of SSER 6); and (c) an exemption from the requirements of paragraph III. B of Option B of 10 CFR Part 50, Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests (SER supporting Amendment 62 to Facility Operating License No. NPF-62). The special circumstances regarding each exemption, except for item (a) above, are identified in the referenced section of the safety evaluation report and the supplements thereto~~.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC Material License No. SNM-1886, issued November 27, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Exelon Generation Company is hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

this exemption

These exemptions 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. ~~The exemptions in items (b) and (c) above are granted pursuant to 10 CFR 50.12.~~ 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. 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], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Clinton Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

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 License Amendment No. 194 and modified by License Amendment No. 206.

Deleted

F. ~~Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report as amended, for the Clinton Power Station, Unit No. 1, and as approved in the Safety Evaluation Report (NUREG-0853) dated February 1982 and Supplement Nos. 1 thru 8 thereto subject to the following provision:~~

> ~~Exelon Generation Company 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.~~

G. Deleted.

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

H. Exelon Generation Company 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.

I. This license is effective as of the date of issuance and ~~shall expire at midnight on September 29, 2026~~.

is effective until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

~~Thomas E. Murley~~, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. ~~Attachments 1 (Deleted) and 2~~
2. Appendix A – Technical Specifications (NUREG-1235)
~~3.~~ Appendix B – Environmental Protection Plan
~~4.~~ ~~Appendix C – Deleted~~

Date of Issuance: ~~April 17, 1987~~

[Insert Date of Issuance]

~~ATTACHMENT 1~~
~~TO NPF-62~~

~~Deleted~~

~~ATTACHMENT 2~~
~~TO NPF-62~~
~~NEW FUEL STORAGE~~

~~Exelon Generation Company shall store new fuel assemblies in accordance with the following requirements.~~

~~a.~~ ~~No more than three fuel assemblies shall be outside their shipping containers, storage racks, or the reactor vessel at any one time.~~

~~b.~~ ~~The minimum edge-to-edge distance between the group of three fuel assemblies and all other fuel assembles shall be 12 inches.~~

~~c.~~ ~~Fuel assemblies, when stored in the New Fuel Storage Vault, shall be stored such that: no more than 12 rows of fuel assembles shall remain uncovered during the loading or unloading of fuel assemblies; metal covers shall cover all other rows containing fuels assemblies during loading and unloading of fuel assemblies; and when loading or unloading of fuel assemblies is not in progress, metal covers shall cover all rows of fuel assemblies.~~

~~d.~~ ~~Fuel assemblies shall be stored in such a manner that water would drain freely from the assemblies in the event of flooding and subsequent draining of the fuel storage area.~~

~~e.~~ ~~Fuel assemblies shall be stored in the containment fuel storage pool only under water.~~

~~f.~~ ~~No fuel assemblies shall be stored in the control rod racks.~~

~~g.~~ ~~All fire hoses servicing the New Fuel Storage Vault shall be equipped with solid stream nozzles.~~

~~April 17, 1987~~

~~APPENDIX C~~

~~ANTITRUST CONDITIONS~~

~~FACILITY OPERATING LICENSE NO. NPF-62~~

~~Deleted~~

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(continued)

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(continued)

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(continued)

1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
~~AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)~~	~~The APLHGR shall be applicable to a specific planar height and is equal to the sum of the LHGRs 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 the height.~~
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provision of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.
~~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 inplace 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.~~
~~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.~~

(continued)

1.1 Definitions (continued)

~~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.~~

~~CORE ALTERATION~~

~~CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:~~

~~a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and~~

~~b. Control rod movement, provided there are no fuel assemblies in the associated core cell.~~

~~Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~

~~CORE OPERATING LIMITS REPORT (COLR)~~

~~The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.~~

~~DOSE EQUIVALENT I-131~~

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same inhalation CEDE dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation CEDE dose conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989.~~

(continued)

1.1 Definitions (continued)

~~EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME~~	~~The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~
~~END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME~~	~~The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~
~~ISOLATION SYSTEM RESPONSE TIME~~	~~The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~

~~(continued)~~

1.1 Definitions (continued)

~~LEAKAGE~~	~~LEAKAGE shall be:~~ ~~a. Identified LEAKAGE~~ ~~1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or~~ ~~2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;~~ ~~b. Unidentified LEAKAGE~~ ~~All LEAKAGE into the drywell that is not identified LEAKAGE;~~ ~~c. Total LEAKAGE~~ ~~Sum of the identified and unidentified LEAKAGE;~~ ~~d. Pressure Boundary LEAKAGE~~ ~~LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.~~
~~LINEAR HEAT GENERATION RATE (LHGR)~~	~~The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.~~
~~LOGIC SYSTEM FUNCTIONAL TEST~~	~~A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.~~

(continued)

1.1 Definitions (continued)

~~MINIMUM CRITICAL POWER RATIO (MCPR)~~	~~The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.~~
~~MODE~~	~~A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.~~
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
~~RATED THERMAL POWER (RTP)~~	~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3473 MWt.~~
~~REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME~~	~~The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~

(continued)

1.1 Definitions (continued)

~~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.~~
~~STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.~~
~~THERMAL POWER~~	~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~
~~TURBINE BYPASS SYSTEM RESPONSE TIME~~	~~The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:~~ ~~a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and~~ ~~b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.~~ ~~The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~

~~Table 1.1-1 (page 1 of 1)~~
~~MODES~~

~~MODE~~	~~TITLE~~	~~REACTOR MODE SWITCH POSITION~~	~~AVERAGE REACTOR COOLANT TEMPERATURE (ºF)~~
~~1~~	~~Power Operation~~	~~Run~~	~~NA~~
~~2~~	~~Startup~~	~~Refuel[(a)] or Startup/Hot Standby~~	~~NA~~
~~3~~	~~Hot Shutdown[(a)]~~	~~Shutdown~~	~~> 200~~
~~4~~	~~Cold Shutdown[(a)]~~	~~Shutdown~~	~~≤ 200~~
~~5~~	~~Refueling[(b)]~~	~~Shutdown or Refuel~~	~~NA~~

~~(a) All reactor vessel head closure bolts fully tensioned.~~

~~(b) One or more reactor vessel head closure bolts less than fully tensioned.~~

1.0 USE AND APPLICATION

1.2 ~~Logical Connectors~~

Deleted

~~PURPOSE~~

~~The purpose of this section is to explain the meaning of logical connectors.~~

~~Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.~~

~~BACKGROUND~~

~~Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.~~

~~When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.~~

~~EXAMPLES~~

~~The following examples illustrate the use of logical connectors.~~

~~(continued)~~

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~ handling and storage of spent nuclear fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., ~~inoperable equipment or~~ variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility ~~unit~~ is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility ~~unit~~ is not within the LCO Applicability.

~~If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.~~

~~Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.~~

(continued)

1.3 Completion Times

DESCRIPTION ~~(continued)~~

~~However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:~~

~~a. Must exist concurrent with the first inoperability; and~~

~~b. Must remain inoperable or not within limits after the first inoperability is resolved.~~

~~The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:~~

~~a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or~~

~~b. The stated Completion Time as measured from discovery of the subsequent inoperability.~~

~~The above Completion Time extension does not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.~~

~~The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES~~

~~The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.~~

~~EXAMPLE 1.3-1~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~12 hours~~ ~~36 hours~~

~~Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.~~

~~The Required Actions of Condition B are to be in MODE 3 within 12 hours AND in MODE 4 within 36 hours. A total of 12 hours is allowed for reaching MODE 3 and a total of 36 hours (not 48 hours) is allowed for reaching MODE 4 from the time that Condition B was entered. If MODE 3 is reached within 6 hours, the time allowed for reaching MODE 4 is the next 30 hours because the total time allowed for reaching MODE 4 is 36 hours.~~

~~If Condition B is entered while in MODE 3, the time allowed for reaching MODE 4 is the next 36 hours.~~

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

(continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-2 (continued)~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.~~

~~On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-3~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One Function X subsystem inoperable.~~	~~A.1 Restore Function X subsystem to OPERABLE status.~~	~~7 days~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
~~B. One Function Y subsystem inoperable.~~	~~B.1 Restore Function Y subsystem to OPERABLE status.~~	~~72 hours~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
~~C. One Function X subsystem inoperable.~~ ~~AND~~ ~~One Function Y subsystem inoperable.~~	~~C.1 Restore Function X subsystem to OPERABLE status.~~ ~~OR~~ ~~C.2 Restore Function Y subsystem to OPERABLE status.~~	~~72 hours~~ ~~72 hours~~

(continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-3 (continued)~~

~~When one Function X subsystem and one Function Y subsystem are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each subsystem, starting from the time each subsystem was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second subsystem was declared inoperable (i.e., the time the situation described in Condition C was discovered).~~

~~If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected subsystem was declared inoperable (i.e., initial entry into Condition A).~~

~~The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-4~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve(s) to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~12 hours~~ ~~36 hours~~

~~A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.~~

~~Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.~~

~~If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-5~~

~~ACTIONS~~

~~-----------------------------NOTE-----------------------------~~
~~Separate Condition entry is allowed for each inoperable valve.~~
~~--~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~12 hours~~ ~~36 hours~~

~~The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.~~

~~The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.~~

(continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-5 (continued)~~

~~If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.~~

~~Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.~~

~~EXAMPLE 1.3-6~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One channel inoperable.~~	~~A.1 Perform SR 3.x.x.x.~~	~~Once per 8 hours~~
	~~OR~~	
	~~A.2 Reduce THERMAL POWER to ≤ 50% RTP.~~	~~8 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~12 hours~~

(continued)

1.3 Completion Times

EXAMPLES

~~EXAMPLE 1.3-6 (continued)~~

~~Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.~~

~~If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-7~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One subsystem inoperable.~~	~~A.1 Verify affected subsystem isolated.~~	~~1 hour~~ ~~AND~~ ~~Once per 8 hours thereafter~~
	~~AND~~	
	~~A.2 Restore subsystem to OPERABLE status.~~	~~72 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~12 hours~~
	~~AND~~	
	~~B.2 Be in MODE 4.~~	~~36 hours~~

~~Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.~~

~~If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after~~

(continued)

1.3 Completion Times

~~EXAMPLES~~	~~EXAMPLE 1.3-7 (continued)~~ ~~Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.~~
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR~~, as well as certain Notes in the Surveillance column that modify performance requirements~~. ~~Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example 1.4-4 discusses these special situations.~~ ~~Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.~~ The use of "met" or "performed" in these instances conveys specified meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance

(continued)

1.4 Frequency

DESCRIPTION (continued)

criteria. ~~SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:~~

~~a. The Surveillance is not required to be performed; and~~

~~b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.~~

EXAMPLES

The following examples ~~illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3. The examples do not reflect the potential application of LCO 3.0.4.b.~~

illustrates the type of frequency statement that appears in the Permanently Defueled Technical Specifications (PDTS).

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

Verify level is within limits

SURVEILLANCE	FREQUENCY
~~Perform CHANNEL CHECK~~.	12 hours

Example 1.4-1 contains the type of SR ~~most often~~ encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when ~~the equipment is inoperable,~~ a variable is outside specified limits, or the facility ~~unit~~ is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the ~~unit~~ facility is in a ~~MODE or other~~ specified condition in the Applicability of the LCO, ~~and the performance of the Surveillance is not otherwise modified (refer to Examples 1.4-3 and 1.4-4),~~ then SR 3.0.3 becomes applicable.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

If the interval as specified by SR 3.0.2 is exceeded while the ~~unit~~ facility is not in a ~~MODE or other~~ specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the ~~MODE or other~~ specified condition. ~~Failure to do so would result in a violation of SR 3.0.4.~~

~~EXAMPLE 1.4-2~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~Verify flow is within limits.~~	~~Once within 12 hours after ≥ 25% RTP~~ ~~AND~~ ~~24 hours thereafter~~

~~Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~

~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2.~~

(continued)

1.4 Frequency

~~EXAMPLES~~

~~EXAMPLE 1.4-2 (continued)~~

~~"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.~~

~~EXAMPLE 1.4-3~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~-------------------NOTE-------------------~~ ~~Not required to be performed until 12 hours after ≥ 25% RTP.~~ ~~--~~ ~~Perform channel adjustment.~~	~~7 days~~

~~The interval continues whether or not the unit operation is < 25% RTP between performances.~~

~~As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.~~

(continued)

1.4 Frequency

~~EXAMPLES~~

~~EXAMPLE 1.4-3 (continued)~~

~~Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.~~

~~EXAMPLE 1.4-4~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~-------------------NOTE-------------------~~ ~~Only required to be met in MODE 1.~~ ~~--~~ ~~Verify leakage rates are within limits.~~	~~24 hours~~

~~Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour (plus the extension allowed by SR 3.0.2) interval, but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.~~

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the ~~MODES or other~~ specified conditions in the Applicability, except as provided in LCO 3.0.2 ~~and LCO 3.0.7~~.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met~~, except as provided in LCO 3.0.5 and LCO 3.0.6~~. 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.
~~LCO 3.0.3~~	~~When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:~~ ~~a. MODE 2 within 7 hours;~~ ~~b. MODE 3 within 13 hours; and~~ ~~c. MODE 4 within 37 hours.~~ ~~Exceptions to this Specification are stated in the individual Specifications.~~ ~~Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.~~ ~~LCO 3.0.3 is only applicable in MODES 1, 2, and 3.~~
~~LCO 3.0.4~~	~~When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:~~ ~~a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;~~

(continued)

3.0 LCO APPLICABILITY

~~LCO 3.0.4 (continued)~~	~~b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or~~ ~~c. When an allowance is stated in the individual value, parameter, or other Specification.~~ ~~This specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~
~~LCO 3.0.5~~	~~Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.~~
~~LCO 3.0.6~~	~~When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.10, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~ ~~When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

(continued)

3.0 LCO APPLICABILITY (continued)

~~LCO 3.0.7~~ ~~Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.~~

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. 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 3.0.3. ~~Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.~~
SR 3.0.2	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~~ or as measured from the time a specified condition of the Frequency is met~~. ~~For Frequencies specified as "once," the above interval extension does not apply.~~ ~~If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.~~ ~~Exceptions to this Specification are stated in the individual Specifications.~~
SR 3.0.3	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.

(continued)

3.0 SR APPLICABILITY (continued)

SR 3.0.3 (continued)	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 3.0.4	Entry into a ~~MODE or other~~ specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. ~~When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

Facility

3.7 ~~PLANT~~ SYSTEMS

3.7.7 Fuel Pool Water Level

LCO 3.7.7 The fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool ~~and upper containment fuel storage pool~~ racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the ~~associated~~ fuel storage pool.

spent

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel pool water level not within limit.	A.1 ~~---------NOTE----------~~ ~~LCO 3.0.3 is not applicable.~~ ~~----------------------~~ Suspend movement of irradiated fuel assemblies in the ~~associated~~ fuel storage pool~~(s)~~. spent	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify the fuel pool water level is ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.	~~In accordance with the Surveillance Frequency Control Program~~ 7 days

4.0 DESIGN FEATURES

4.1 Site Location

The site for the Clinton Power Station is located in Harp Township, DeWitt County, approximately six miles east of the city of Clinton in east-central Illinois. The exclusion area boundary shall have a radius of 975 meters from the Standby Gas Treatment System vent.

4.2 ~~Reactor Core~~

Deleted

~~4.2.1 Fuel Assemblies~~

~~The reactor shall contain 624 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material, and water rod(s). Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.~~

~~A maximum of eight GE14i isotope test assemblies will be placed in non-limiting core regions, beginning with the Reload 12 Cycle 13 core reload, with the purpose of obtaining surveillance data to verify that the GE14i assemblies perform satisfactorily in service prior to use of these design features on a production basis. Each GE14i assembly contains a small number of zircaloy-2 clad isotope rods that contain Cobalt-59 targets. These Cobalt-59 targets will transition into Cobalt-60 isotope targets during the cycle irradiation of the assemblies. Details of the GE14i assemblies are contained in NEDC-33505P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Clinton Power Station," Revision 0, dated June 2009.~~

~~4.2.2 Control Rod Assemblies~~

~~The reactor core shall contain 145 cruciform shaped control rod assemblies. The control material shall be boron carbide or hafnium metal, or both.~~

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the USAR;

b. ~~A nominal fuel assembly center to center storage spacing of 7 inches within rows and 12.25 inches between rows in the low density storage racks in the upper containment pool;~~ and

~~c.~~ For the fuel storage racks supplied by Nuclear Energy Services (NES), a nominal fuel assembly spacing of 6.4375 inches in the high density storage racks in the spent fuel pool or fuel cask storage pool. For the fuel storage racks supplied by Holtec International, a nominal fuel assembly spacing of 6.243 inches in the high density storage racks in the spent fuel pool or fuel cask storage pool.

~~4.3.1.2 The new fuel storage racks are designed and shall be maintained with:~~

~~a.~~ ~~$k_{eff} \leq$ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the USAR; and~~

~~b. A nominal fuel assembly center to center storage spacing of 7 inches within rows and 12.25 inches between rows in the new fuel storage racks.~~

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 754 ft 0 inches.

4.3.3 Capacity

4.3.3.1 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3796 fuel assemblies. The fuel cask storage pool is designed and shall be maintained with a storage capacity limited to no more than 363 fuel assemblies.

~~4.3.3.2 No more than 160 fuel assemblies may be stored in the upper containment pool.~~

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall ~~unit~~ facility operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager, or his designee, shall approve, prior to implementation, each proposed test, experiment, and modification to systems or equipment that affect ~~nuclear safety~~ safe storage and maintenance of spent nuclear fuel.

5.1.2 The shift supervisor (SS) shall be responsible for the ~~control room~~ shift command function. ~~During any absence of the SS from the control room while the unit is in MODE 1, 2, or 3, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 4 or 5, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.~~

NOTE: Proposed changes to TS Section 5.0, with the exception of page renumbering, are under NRC review. Reference: Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition," dated July 28, 2016 (ML16210A300)

NOTE: Proposed changes to TS Section 5.0, with the exception of page renumbering, are under NRC review. Reference: Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition - Supplement 1," dated November 4, 2016 (ML16309A013)

5.2.1 Onsite and Offsite Organizations

facility staff

Onsite and offsite organizations shall be established for ~~unit operation~~ and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting ~~safety of the nuclear power plant~~.

safe storage and handling of spent nuclear fuel

facility

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all ~~operating~~ organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the ~~plant~~ specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the USAR;

facility

facility

responsible officer

the safe storage and handling of spent nuclear fuel

b. The plant manager shall be responsible for overall safe operation of the ~~plant~~ and shall have control over those onsite activities necessary for safe ~~operation~~ and maintenance of ~~the plant~~;

spent nuclear fuel

storage

facility to ensure safe management of spent nuclear fuel

c. A ~~specified corporate executive~~ shall have corporate responsibility for ~~overall plant nuclear safety~~ and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the ~~plant to ensure nuclear safety~~; and

CERTIFIED FUEL HANDLERs

d. The individuals who train the ~~operating staff~~, carry out radiation protection, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ~~independence from operating pressures~~.

ability to perform their assigned functions

5.2.2 ~~Unit~~ Staff

Facility

The ~~unit~~ staff organization shall include the following:

facility

a. ~~A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be on site while the unit is in MODE 1, 2, or 3.~~

(continued)

Each duty shift shall be composed of at least one shift supervisor and one NON-CERTIFIED OPERATOR. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

5.2 Organization

Facility

5.2.2 ~~Unit~~ Staff (continued)

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.

b. ~~At least one licensed RO shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed SRO shall be present in the control room.~~

c. Shift crew composition may be one less than the minimum requirements of ~~10 CFR 50.54(m)(2)(i) and~~ Specifications 5.2.2.a ~~and 5.2.2.g~~ for a period of time not to exceed 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition within the minimum requirements.

during the movement of fuel and during the movement of loads over fuel

d. A radiation protection technician shall be on site ~~when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.~~

e. ~~Deleted.~~ Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

f. The shift supervisor shall be a CERTIFIED FUEL HANDLER. ~~The operations manager or at least one operations middle manager shall hold an SRO license for Clinton Power Station.~~

g. Deleted ~~The Shift Technical Advisor (STA) shall provide advisory technical support to the SS in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~

and the following conditions are met:
1) No fuel movement is in progress;
2) No movement of loads over the spent fuel is in progress.
This provision does not permit any shift crew position to be unstaffed upon shift change due to the absence or tardiness of an oncoming shift crew member

NOTE: Proposed changes to TS Section 5.0, with the exception of page renumbering, are under NRC review. Reference: Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition - Supplement 1," dated November 4, 2016 (ML16309A013)

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Facility

Facility

facility

5.0 ADMINISTRATIVE CONTROLS

5.3 ~~Unit~~ Staff Qualifications

5.3.1 Each member of the ~~unit~~ staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978~~, with the following exception: the licensed operators who shall comply only with the requirements of 10 CFR 55.~~

5.3.2 An NRC-approved training and retraining program for the CERTIFIED FUEL HANDLER shall be maintained.

for comparable positions with exceptions specified in the Quality Assurance Program Manual.

NOTE: Proposed changes to TS Section 5.0, with the exception of page renumbering and the clouded statement, are under NRC review. Reference: Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Proposed Changes to Technical Specifications Section 5.0 Administrative Controls for Permanently Defueled Condition," dated July 28, 2016 (ML16210A300)

4

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

applicable to safe storage of nuclear fuel

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The ~~applicable~~ procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

Deleted

~~b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;~~

c. Quality assurance for effluent and environmental monitoring; and

d. All programs specified in Specification 5.5.

5

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 ~~Primary Coolant Sources Outside Containment~~ Deleted

~~This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the:~~

~~a. LPCS System;~~

~~b. HPCS System;~~

~~c. RHR System;~~

~~d. RCIC System;~~

~~e. Suppression Pool Makeup System;~~

~~f. Combustible Gas Control System;~~

~~g. Containment Monitoring System; and~~

~~h. Post-accident Sampling System (until such time as a modification eliminates the PASS penetration as a potential leakage path).~~

~~The program shall include the following:~~

~~a. Preventive maintenance and periodic visual inspection requirements;~~

~~b. Integrated leak test requirements for each system at least once per 24 months; and~~

~~c. In the event work is performed which could result in leakage from a component or system covered by this program, a visual inspection shall be performed and repairs made as required.~~

~~The specified frequency for integrated leak testing is met if the testing is performed within 1.25 times the interval specified.~~

(continued)

5.5 Programs and Manuals (continued)

5.5.3 Deleted

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the ~~unit~~ facility to unrestricted areas, conforming to 10 CFR 50, Appendix I;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

e. 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;

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

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary as follows:

1. For noble gases: ≤ 500 mrem/yr to the total body and ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days: ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the ~~unit~~ facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the ~~unit~~ facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

(continued)

5.5 Programs and Manuals (continued)

5.5.5 ~~Component Cyclic or Transient Limit~~ Deleted

~~This program provides controls to track the cyclic and transient occurrences identified on USAR Table 3.9-1(b) to ensure that the reactor vessel is maintained within the design limits.~~

5.5.6 ~~Inservice Testing Program~~ Deleted

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:~~

~~a.~~ ~~Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days;~~

~~b.~~ ~~The provisions of SR 3.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c.~~ ~~The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

~~d.~~ ~~Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.~~

(continued)

5.5 Programs and Manuals (continued)

5.5.7 ~~Ventilation Filter Testing Program (VFTP)~~ Deleted

~~A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, except that testing specified at a frequency of 18 months is required at a frequency of 24 months.~~

~~a.~~ ~~Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:~~

~~ESF Ventilation System~~	~~Flowrate~~
~~SGTS~~	~~4,000 cfm~~
~~Control Room Ventilation (CRV) Makeup Filter~~	~~3,000 cfm~~

~~b.~~ ~~Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:~~

~~ESF Ventilation System~~	~~Flowrate~~	~~Penetration and Bypass~~
~~SGTS~~	~~4,000 cfm~~	~~0.05%~~
~~CRV Makeup Filter~~	~~3,000 cfm~~	~~0.05%~~
~~CRV Recirculation Filter~~	~~64,000 cfm~~	~~2%~~

~~c.~~ ~~Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30 ºC and a relative humidity of 70%:~~

~~ESF Ventilation System~~	~~Penetration~~
~~SGTS~~	~~0.175%~~
~~CRV Makeup Filter~~	~~0.175%~~
~~CRV Recirculation Filter~~	~~6%~~

(continued)

5.5 Programs and Manuals

~~5.5.7~~ ~~Ventilation Filter Testing Program (VFTP) (continued)~~

~~d.~~ ~~Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 6.0 inches water gauge when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:~~

~~ESF Ventilation System~~	~~Flowrate~~
~~SGTS~~	~~4,000 cfm~~
~~CRV Makeup Filter~~	~~3,000 cfm~~

~~e.~~ ~~Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1980:~~

~~ESF Ventilation System~~	~~Wattage~~
~~SGTS~~	~~18.0 kW~~
~~CRV Makeup Filter~~	~~14.4 kW~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.~~

5.5.8 ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program

This program provides controls for ~~potentially explosive gas mixtures contained in the main condenser offgas treatment system and~~ the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. ~~The limits for concentrations of hydrogen in the main condenser offgas treatment system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and~~

(continued)

5.5 Programs and Manuals

5.5.8 ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program (continued)

~~b.~~ A surveillance program to ensure that the quantity of radioactive material contained in each outdoor tank that is not surrounded by a liner, dike, or walls capable of holding the tank's contents and that does not have a tank overflow and surrounding area drains connected to the liquid radwaste treatment system is ≤ 10 curies, excluding tritium and dissolved or entrained noble gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.9 ~~Diesel Fuel Oil Testing Program~~ Deleted

~~A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:~~

~~a.~~ ~~Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:~~

~~1.~~ ~~an API gravity or an absolute specific gravity within limits,~~

~~2.~~ ~~a kinematic viscosity within limits for ASTM 2D fuel oil, and~~

~~3.~~ ~~a water and sediment content within limits or a clear and bright appearance with proper color;~~

~~b.~~ ~~Other properties of the new fuel oil are within limits for ASTM 2D fuel oil within 31 days of addition to the storage tanks; and~~

~~c.~~ ~~Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days.~~

(continued)

5.5 Programs and Manuals (continued)

5.5.10 ~~Safety Function Determination Program (SFDP)~~ Deleted

~~This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:~~

~~a. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;~~

~~b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;~~

~~c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and~~

~~d. Other appropriate limitations and remedial or compensatory actions.~~

~~A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:~~

~~a. A required system redundant to system(s) supported by the inoperable support system is also inoperable;~~

~~b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or~~

~~c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.~~

~~The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

(continued)

5.5 Programs and Manuals (continued)

5.5.11 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 USAR 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 USAR.

d. Proposed changes that meet the criteria of either Specification 5.5.11.b.1 or Specification 5.5.11.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).

~~5.5.12 Ultimate Heat Sink (UHS) Erosion, Sediment Monitoring, and Dredging Program~~

~~A program to provide maintenance on the UHS in the event inspections of the UHS dam, its abutments, or the UHS shoreline indicate erosion or local instability. This program shall ensure that the UHS is maintained in such a way as to achieve the following objectives:~~

~~a. During normal operation, there will be a volume of water in the UHS below elevation 675 sufficient to receive the sediment load from a once-in-25-year flood event; and~~

~~b. Still be adequate to maintain the plant in a safe-shutdown condition for 30 days under meteorological conditions of the severity suggested by Regulatory Guide 1.27.~~

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995,as modified by the following exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests, and (2) NEI 94-01-1995, Section 9.2.3: The first Type A test performed after November 23, 1993 shall be performed no later than November 23, 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 9.0 psig.

The maximum allowable primary containment leakage rate L_a, at P_a, shall be 0.65% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq$ 5 scfh when tested at $\geq P_a$,

2) For each door, leakage rate is $\leq$ 5 scfh when the gap between door seals is pressurized $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

5.5 Program and Manuals (continued)

~~5.5.14~~ ~~Battery Monitoring and Maintenance Program~~

~~This program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications," including the following:~~

~~a.~~ ~~Actions to restore battery cells with float voltage < 2.13 V,~~

~~and~~

~~b.~~ ~~Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.~~

~~5.5.15~~ ~~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 Ventilation 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 5 rem total effective dose equivalent (TEDE) for the duration of the accident. 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 Section 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 Section C.1 and C.2 of Regulatory Guide 1.197, Revision 0.~~

(continued)

5.5 Program and Manuals (continued)

~~5.5.15~~ ~~Control Room Envelope Habitability Program (continued)~~

~~d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem of the Control Room Ventilation System, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as a part of the 24 month assessment of the CRE boundary.~~

~~e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. 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 SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraph c and d, respectively.~~

~~5.5.16~~ ~~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 Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.~~

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Deleted

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~ facility during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses 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 the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the ~~unit~~ facility during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the ~~unit~~ 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 50, Appendix I, Section IV.B.1.

~~5.6.4~~ ~~Deleted~~

(continued)

5.6 Reporting Requirements

~~5.6.5~~ ~~CORE OPERATING LIMITS REPORT (COLR)~~

~~a.~~ ~~Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:~~

~~1.~~ ~~LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),~~

~~2.~~ ~~LCO 3.2.2, Minimum Critical Power Ratio (MCPR),~~

~~3.~~ ~~LCO 3.2.3, Linear Heat Generation Rate (LHGR),~~

~~4.~~ ~~LCO 3.3.1.1, RPS Instrumentation (SR 3.3.1.1.14),~~

~~5.~~ ~~LCO 3.3.1.3, Oscillation Power Range Monitor (OPRM) Instrumentation, and~~

~~6.~~ ~~LCO 3.7.6, Main Turbine Bypass System, (cycle dependent thermal power limits for an inoperable Main Turbine Bypass System).~~

~~b.~~ ~~The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in~~

~~(1)~~ ~~General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A, or~~

~~(2)~~ ~~NEDO-32465, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications."~~

~~c.~~ ~~The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~

~~d.~~ ~~The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.~~

(continued)

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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.~~7~~ 2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	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 ~~unit~~ facility is in the ~~MODES or other~~ specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	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. ~~There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the~~

(continued)

BASES

LCO 3.0.2 (continued)

~~unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.~~

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

~~The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.11, "RCS Pressure and Temperature (P/T) Limits."~~

~~The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/divisions of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.~~

~~When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.~~

(continued)

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 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. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

~~Systems and components are assumed to be OPERABLE when the associated SRs 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 SRs; 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~~ facility is in a ~~MODE or other~~ specified condition for which the requirements of the associated LCO are not applicable~~, unless otherwise specified~~. ~~The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.~~

~~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 in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.~~

(continued)

BASES

SR 3.0.1 ~~(continued)~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. 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. Some examples of this process are:~~

~~a.~~ ~~Control rod drive maintenance during refueling that requires scram testing at ≥ 950 psig. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 950 psig to perform other necessary testing.~~

~~b.~~ ~~Reactor Core Isolation Cooling (RCIC) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with RCIC considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.~~

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances~~ and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval~~.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

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BASES

SR 3.0.2 (continued)

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 exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the TS will then include a Note stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Primary Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.~~

~~As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.~~

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals ~~(other than those consistent with refueling intervals) or periodic Completion Time intervals~~ beyond those specified.

SR 3.0.3

SR 3.0.3 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 SR 3.0.2, and not at the time

(continued)

BASES

SR 3.0.3 (continued)

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.

facility

The basis for this delay period includes consideration of ~~unit~~ 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, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 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, SR 3.0.3 allows for 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. SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.~~

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 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~~ 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 action thresholds, and risk management actions up to an including plant shutdown. The~~

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(continued)

BASES

SR 3.0.3 ~~(continued)~~

~~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 be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk 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 Clinton Power Station Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the ~~equipment is considered inoperable or the~~ variable then 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 SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a ~~MODE or other~~ specified condition in the Applicability.

variable limits

facility

This Specification ensures that ~~system and component OPERABILITY requirements and~~ variable limits are met before entry into ~~MODES or other~~ specified conditions in the Applicability for which these ~~systems and components~~ ensure safe operation of the ~~unit~~. ~~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 MODE or other specified condition in the Applicability.~~

~~A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to Surveillance not being met in accordance with LCO 3.0.4.~~

(continued)

BASES

SR ~~3.0.4 (continued)~~

~~However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE 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 SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 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 Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.~~

~~The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES 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 MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2 or 3, MODE 2 to MODE 3, and MODE 3 to MODE 4.~~

~~The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.~~

Facility

B 3.7 ~~PLANT~~ SYSTEMS

B 3.7.7 Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool ~~and upper containment fuel storage pool~~ meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool ~~and upper containment fuel storage pool design~~ is found in the USAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section 15.7.4, Ref. 3) of the 10 CFR 100 (Ref. 4) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto stored fuel bundles. The consequences of a fuel handling accident inside the fuel building ~~and inside containment~~ are documented in Reference 2. The water levels in the spent fuel storage pool ~~and upper containment fuel storage pool~~ provide 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 ~~primary or~~ secondary containment atmosphere~~, as applicable.~~ This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The fuel pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The specified water level preserves the assumption of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool ~~and upper containment fuel storage pool~~.

APPLICABILITY

Spent

This LCO applies whenever movement of irradiated fuel assemblies occurs in the ~~associated~~ fuel storage racks since the potential for a release of fission products exists.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.~~

the spent fuel storage

spent fuel

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. With ~~either fuel~~ pool level less than required, the movement of irradiated fuel assemblies in the ~~associated~~ storage pool is suspended immediately. 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 associated fuel storage pool.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool ~~and upper containment fuel storage pool~~ must be checked periodically. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

The 7 day frequency 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.

With regard to fuel pool water level values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 6).

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 9.1.2.
2. USAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4, Revision 1, July 1981.
4. 10 CFR 100.
5. Regulatory Guide 1.25, March 1972.
6. Calculation IP-0-0105.

7. Calculation IP-F-0179