Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
Web Address: www.dom.com

Dominion®

May 29, 2013

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 13-226
LIC/JG/R0
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 256, PERMANENTLY DEFUELED LICENSE AND TECHNICAL SPECIFICATIONS

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would revise the Operating License and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation.

On February 25, 2013, DEK submitted a certification to the NRC indicating its intention to permanently cease power operations at KPS (Reference 1) pursuant to 10 CFR 50.82(a)(1)(i). The certification stated that DEK had decided to permanently cease power operation of KPS on May 7, 2013. On February 26, 2013, DEK submitted both a Post-Shutdown Decommissioning Activities Report (PSDAR) (Reference 2) and an updated Irradiated Fuel Management Plan (Reference 3). With the docketing of the subsequent certification for permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii) on May 14, 2013 (Reference 4), the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). In support of this condition, the KPS license and associated TS are being proposed for revision to comport to this permanently shutdown and defueled condition in accordance with 10 CFR 50.36(c)(6).

Attachment 1 to this letter contains a description, technical analysis, significant hazards determination, and environmental considerations evaluation for the proposed amendment. Attachment 2 contains marked-up 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 marked-up TS Bases pages (for information).

As discussed in this submittal, most of the design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building. A description of the FHA analysis for the permanently defueled condition that was incorporated into the KPS USAR under 50.59 is included for information.

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NRR

The KPS Facility Safety Review Committee has reviewed the proposed amendment and a copy of this submittal has been provided to the State of Wisconsin in accordance with 10 CFR 50.91(b).

DEK requests approval of the proposed amendment by June 1, 2014. Once approved, the amendment shall be implemented within 90 days.

Please contact Mr. Jack Gadzala at 920-388-8604 if you have any questions or require additional information.

Sincerely,

Eugene S. Grecheck
Vice President – Nuclear Engineering and Development

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President – Nuclear Engineering and Development of Dominion Energy Kewaunee, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 29TH day of MAY, 2013.

My Commission Expires: MAY 31, 2014.

Vicki L. Hull
Notary Public

Attachments:

1. Discussion of Change, Technical Analysis, Significant Hazards Determination and Environmental Considerations
2. Marked-up Technical Specifications Pages
3. Marked-up Technical Specifications Bases Pages

References:

1. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Certification of Permanent Cessation of Power Operations," dated February 25, 2013 (ADAMS Accession No. ML13058A065).
2. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Post-Shutdown Decommissioning Activities Report," dated February 26, 2013 (ADAMS Accession No. ML13063A248).
3. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Update to Irradiated Fuel Management Plan Pursuant to 10 CFR 50.54(bb)," dated February 26, 2013 (ADAMS Accession No. ML13059A028).
4. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Certification of Permanent Removal of Fuel from the Reactor Vessel," dated May 14, 2013 (ADAMS Accession No. ML13135A209).

cc: Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Mr. K. D. Feintuch
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop O8-D15
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Kewaunee Power Station

Public Service Commission of Wisconsin
Electric Division
P.O. Box 7854
Madison, WI 53707

ATTACHMENT 1

LICENSE AMENDMENT REQUEST 256
PERMANENTLY DEFUELED LICENSE AND TECHNICAL SPECIFICATIONS

DISCUSSION OF CHANGE, TECHNICAL ANALYSIS, SIGNIFICANT HAZARDS DETERMINATION, AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

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PERMANENTLY DEFUELED LICENSE AND TECHNICAL SPECIFICATIONS

DISCUSSION OF CHANGE, TECHNICAL ANALYSIS, SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL CONSIDERATIONS

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would revise the Operating License and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the permanent cessation of reactor operation.

On February 25, 2013, DEK submitted a certification to the NRC indicating its intention to permanently cease power operations at KPS (Reference 1) pursuant to 10 CFR 50.82(a)(1)(i). The certification stated that DEK had decided to permanently cease power operation of KPS on May 7, 2013. On February 26, 2013, DEK submitted both a Post-Shutdown Decommissioning Activities Report (PSDAR) (Reference 2) and an updated Irradiated Fuel Management Plan (Reference 3). With the docketing of the subsequent certification for permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii) on May 14, 2013, the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). In support of this condition, the KPS license and associated TS are being proposed for revision to comport to this permanently shutdown and defueled condition in accordance with 10 CFR 50.36(c)(6).

The existing KPS 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. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a 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, as described in the KPS safety analyses.

On April 16, 2013, DEK submitted License Amendment Request LAR 255 (Reference 8) to delete KPS Renewed Facility Operating License Condition 2.C.(15), "License Renewal License Conditions." For completeness, the proposed amendment being submitted herein (LAR 256) also includes the proposed changes in LAR 255 in their entirety with no alteration. Approval of LAR 255 continues to be requested prior to December 22, 2013, as discussed in that submittal. Upon approval of LAR 255, the corresponding change to License Condition 2.C.(15) in LAR 256 would thus be unnecessary.

There are no other pending license amendment requests currently docketed for KPS. Therefore, no disposition of other TS changes, as they relate to this license amendment request, is needed.

2.0 PROPOSED CHANGE

The proposed amendment would modify the KPS license to comport to a permanently defueled condition and revise KPS Technical Specifications (TS) into Permanently Defueled Technical Specifications (PDTS).

General Analysis Applicable to Proposed Change

Section 14 of the KPS Updated Safety Analysis Report (USAR) described the design basis accident (DBA) and transient scenarios applicable to KPS during power operations. During normal power operations, the forced flow of water through the reactor coolant system (RCS) removes the heat generated by the reactor. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator tubes to the secondary system. 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 systems which could affect the reactor core.

However, as a result of the certifications submitted by DEK 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), most of the accident scenarios postulated in the USAR are no longer possible.

DEK plans to use a decommissioning method called SAFSTOR, in which most fluid systems are drained and the plant is left in a stable condition until final dismantlement. The irradiated fuel will be stored in the spent fuel pool (SFP) and in the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped off site in accordance with the schedules described in the PSDAR and updated Irradiated Fuel Management Plan. During decommissioning, the spent fuel pool and its systems will be isolated and dedicated only to spent fuel storage. In this condition the spectrum of credible accidents is much smaller than for an operational plant.

A list of the USAR Chapter 14 DBAs is provided in Section 5.2, "Applicable Regulatory Requirements/Criteria," of this submittal. Of these, the only accident scenarios that could potentially apply to a permanently defueled facility would be a fuel handling accident (FHA), an accidental release of waste liquid, or an accidental release of waste gas. Since the waste gas decay tanks, volume control tanks, liquid holdup tanks, reactor coolant drain tank, and associated systems will be purged of their contents, a rupture of these components would no longer be an applicable initiator or source of such an accident. The analyzed accident that remains applicable to KPS in the permanently shut down and defueled condition is a FHA in the auxiliary building where the SFP is located. The FHA analysis for KPS shows that, following 90 days of decay time after reactor shutdown and provided the spent fuel pool water level requirements of TS 3.7.13 are met[1], the dose consequences are acceptable without relying on structures, systems, and components (SSCs) remaining functional for accident mitigation during and following the event. (The one exception to this is the continued function of the passive spent fuel pool structure).

The definition of safety-related structures, systems, and components (SSCs) in 10 CFR 50.2, "Definitions," states that safety-related SSCs are those relied on to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant boundary;
2. The capability to shutdown 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 10 CFR 50.43(a)(1) or 100.11.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after the termination of reactor operations at KPS and the permanent removal of the fuel from the reactor vessel (following 90 days of decay time after shutdown), none of the SSCs at KPS are required to be relied upon for accident mitigation. Therefore, none of the SSCs at KPS meet the definition of a safety-related SSC stated in 10 CFR 50.2 (with the exception of the passive spent fuel pool structure).

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 KPS TS is limited to those needed to address the remaining applicable design basis accident

[1] KPS TS 3.7.13, "Spent Fuel Pool Water Level," requires the spent fuel pool water level to be ≥ 23 feet above the top of irradiated fuel assemblies seated in the storage racks. TS 3.7.13 is applicable during movement of irradiated fuel assemblies in the spent fuel pool.

(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

A fuel handling accident (FHA) was incorporated into the KPS USAR under the provisions of 10 CFR 50.59 to address the permanently defueled condition. The analysis determined a reasonable time post-cessation of operations for movement of fuel from the spent fuel pool during which, if a fuel handling accident occurs, dose consequences would be within 10 CFR 50.67 and Regulatory Guide 1.183 dose limits, given spent fuel pool decontamination based on 23 feet of water over the failed fuel assembly, no credit for emergency ventilation or filtration (control room or otherwise) and no credit for control room atmospheric dispersion for a bounding upper limit of acceptable control room unfiltered inflow.

The FHA is defined as the dropping of a spent fuel assembly onto the spent fuel pool floor or the racks that hold the spent fuel such that the cladding of all the fuel rods in one assembly ruptures. This accident is postulated to occur despite the administrative controls and physical limitations imposed on fuel-handling operations. The gap activity in the damaged rods is instantaneously released into the spent fuel pool. The release occurs under 23 ft of water, which acts as a filter. The activity released from the spent fuel pool then mixes with the auxiliary building atmosphere before being exhausted to the environment. The auxiliary building exhaust rate is established to complete the release in 2 hours consistent with RG 1.183.

The following FHA dose results are associated with a single damaged fuel assembly with 90 days post-cessation of operations decay:

	Dose Limits	90-Day Decayed Dose
CR – Control Room	**5.0 Rem**	**1.9 Rem**
EAB – Exclusion Area Boundary	**6.3 Rem**	**0.001 Rem**
LPZ – Low Population Zone	**6.3 Rem**	**0.001 Rem**

The new analysis differs from the current analysis of record (AOR) with respect to the following model parameters:

	Current AOR	New Analysis
ST – Source Term Decay	**100 hours**	**90 days**
CR – Control Room:		
• **Isolation**	**Credited**	**Not Credited**
• **Recirculation**	**Credited**	**Not Credited**
• **Unfiltered Inflow****	**Pre & Post Isolation***	**3,000 cfm****
X/Q – Atmospheric Dispersion	**Credited**	**Not Credited**

* Pre-Isolation is unfiltered normal intake. Post–Isolation is unfiltered inleakage.
** Combined unfiltered normal intake & unfiltered inleakage for entire accident duration, which was varied between 400 – 6,000 cfm to maximize dose.

The following offsite atmospheric dispersion factors were used for the two-hour release:

EAB	**2.232 E-4 sec/m^3**
LPZ	**3.977 E-5 sec/m^3**

No credit is taken for control room isolation based on the radiation monitor in the control room HVAC duct. Furthermore, no control room isolation or recirculation filtration is assumed in the analysis of post-cessation of operations. All activity is postulated to be released from the spent fuel pool within 2 hours.

Atmospheric dispersion is not credited in the determination of control room dose, which models the source and receptor as collocated using a value of 1. This is conservative because the actual radiological plume will experience some dispersion in the environment in route to the control room intake. All fuel is postulated to have been removed from containment after 100 hours but prior to 90 days after permanent shutdown.[2] The current licensing basis FHA considers both the spent fuel pool and containment as release sources; however, only the spent fuel pool is assumed as a release source after 90 days post-cessation of operations. The FHA assumes one fuel assembly is dropped into the spent fuel pool and all fuel rods in that assembly are damaged release all their gap activity. The FHA analysis postulates that release to the spent fuel pool atmosphere is not mitigated en route to the environment. This assumption is consistent with the current licensing basis FHA analysis, which does not credit the spent fuel pool ventilation system (in auxiliary building) for accident mitigation.

[2] The KPS reactor was permanently shut down on May 7, 2013. Defueling of the reactor started on May 12, 2013 and was completed on May 14, 2013.

2.1 Technical Specifications

The following table provides a summary of which TS are being deleted in their entirety and which TS are being revised into the PDTS. The details of, and justification for the proposed changes follow in subsequent sections (arranged by TS Section).

TS Being Deleted	TS Being Revised
1.0 USE AND APPLICATION	
	1.1 Definitions
	1.2 Logical Connectors
	1.3 Completion Times
	1.4 Frequency
2.0 SAFETY LIMITS (SLs)	
2.1 Safety Limits (SLs)	
2.2 SL Violations	
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	
	3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
	3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1 REACTIVITY CONTROL SYSTEMS	
3.1.1 SHUTDOWN MARGIN (SDM)	
3.1.2 Core Reactivity	
3.1.3 Moderator Temperature Coefficient (MTC)	
3.1.4 Rod Group Alignment Limits	
3.1.5 Shutdown Bank Insertion Limits	
3.1.6 Control Bank Insertion Limits	
3.1.7 Rod Position Indication	
3.1.8 PHYSICS TESTS Exceptions - MODE 2	

TS Being Deleted		TS Being Revised
3.2	**POWER DISTRIBUTION LIMITS**	
3.2.1	Heat Flux Hot Channel Factor (FQ(Z))	
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	
3.2.3	AXIAL FLUX DIFFERENCE (AFD)	
3.2.4	QUADRANT POWER TILT RATIO (QPTR)	
3.3	**INSTRUMENTATION**	
3.3.1	Reactor Protection System (RPS) Instrumentation	
3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	
3.3.3	Post Accident Monitoring (PAM) Instrumentation	
3.3.4	Dedicated Shutdown System	
3.3.5	Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation	
3.3.6	Containment Purge and Vent Isolation Instrumentation	
3.3.7	Control Room Post Accident Recirculation (CRPAR) System Actuation Instrumentation	
3.4	**REACTOR COOLANT SYSTEM (RCS)**	
3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	
3.4.2	RCS Minimum Temperature for Criticality	
3.4.3	RCS Pressure and Temperature (P/T) Limits	
3.4.4	RCS Loops - MODES 1 and 2	
3.4.5	RCS Loops - MODE 3	
3.4.6	RCS Loops - MODE 4	
3.4.7	RCS Loops - MODE 5, Loops Filled	
3.4.8	RCS Loops - MODE 5, Loops Not Filled	
3.4.9	Pressurizer	
3.4.10	Pressurizer Safety Valves	

TS Being Deleted		TS Being Revised
3.4.11	Pressurizer Power Operated Relief Valves (PORVs)	
3.4.12	Low Temperature Overpressure Protection (LTOP) System	
3.4.13	RCS Operational LEAKAGE	
3.4.14	RCS Pressure Isolation Valve (PIV) Leakage	
3.4.15	RCS Leakage Detection Instrumentation	
3.4.16	RCS Specific Activity	
3.4.17	Steam Generator (SG) Tube Integrity	
3.5	**EMERGENCY CORE COOLING SYSTEMS (ECCS)**	
3.5.1	Accumulators	
3.5.2	ECCS - Operating	
3.5.3	ECCS - Shutdown	
3.5.4	Refueling Water Storage Tank (RWST)	
3.6	**CONTAINMENT SYSTEMS**	
3.6.1	Containment	
3.6.2	Containment Air Locks	
3.6.3	Containment Isolation Valves	
3.6.4	Containment Pressure	
3.6.5	Containment Air Temperature	
3.6.6	Containment Spray and Cooling Systems	
3.6.7	Spray Additive System	
3.6.8	Shield Building	
3.6.9	Vacuum Relief Valves	
3.6.10	Shield Building Ventilation System (SBVS)	
3.7	**PLANT SYSTEMS**	
3.7.1	Main Steam Safety Valves (MSSVs)	
3.7.2	Main Steam Isolation Valves (MSIVs)	
3.7.3	Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs), and MFRV Bypass Valves	
3.7.4	Steam Generator (SG) Power Operated Relief Valves (PORVs)	

TS Being Deleted		TS Being Revised	
3.7.5	Auxiliary Feedwater (AFW) System		
3.7.6	Condensate Storage Tanks (CSTs)		
3.7.7	Component Cooling (CC) System		
3.7.8	Service Water (SW) System		
3.7.9	Ultimate Heat Sink (UHS)		
3.7.10	Control Room Post Accident Recirculation (CRPAR) System		
3.7.11	Control Room Air Conditioning (CRAC) Alternate Cooling System		
3.7.12	Auxiliary Building Special Ventilation (ASV) System		
		3.7.13	Spent Fuel Pool Water Level
		3.7.14	Spent Fuel Pool Boron Concentration
		3.7.15	Spent Fuel Pool Storage
3.7.16	Secondary Specific Activity		
3.8	**ELECTRICAL POWER SYSTEMS**		
3.8.1	AC Sources - Operating		
3.8.2	AC Sources - Shutdown		
3.8.3	Diesel Fuel Oil and Lube Oil		
3.8.4	DC Sources - Operating		
3.8.5	DC Sources - Shutdown		
3.8.6	Battery Parameters		
3.8.7	Inverters - Operating		
3.8.8	Inverters - Shutdown		
3.8.9	Distribution Systems - Operating		
3.8.10	Distribution Systems - Shutdown		
3.9	**REFUELING OPERATIONS**		
3.9.1	Boron Concentration		
3.9.2	Nuclear Instrumentation		
3.9.3	Residual Heat Removal (RHR) and Coolant Circulation – High Water Level		
3.9.4	Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level		
3.9.5	Refueling Cavity Water Level		
3.9.6	Containment Penetrations		

TS Being Deleted	TS Being Revised
4.0 DESIGN FEATURES	
	4.1 Site Location
4.2 Reactor Core	
	4.3 Fuel Storage
5.0 ADMINISTRATIVE CONTROLS	
	5.1 Responsibility
	5.2 Organization
	5.3 Unit Staff Qualifications
	5.4 Procedures
	5.5 Programs and Manuals
	5.6 Reporting Requirements
	5.7 High Radiation Area

The TS Table of Contents is being revised accordingly.

The corresponding TS Bases are also being either deleted or revised (as applicable) to reflect these changes.

2.2 Renewed Facility Operating License

This section describes the proposed changes to the KPS Renewed Facility Operating License and the justification for each change.

License Condition 1.B

B. ~~Construction of the Kewaunee Power Station (facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-50, as amended, and the application, as amended, the provisions of the Act and the rules and regulations of the Commission~~

This section is proposed for deletion in its entirety. Decommissioning of KPS is not dependent on the regulations that governed construction of the facility.

This paragraph will read as follows.

B. Deleted;

License Condition 1.I

I. ~~The receipt, possession, and use of byproduct, source, and special nuclear material as authorized by this renewed operating license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including 10 CFR Sections 30.33, 70.23, and 70.31~~

This section is proposed for deletion in its entirety. The Commission's finding regarding possession and use of byproduct, source, and special nuclear material for operating KPS is not dependent on decommissioning of the facility. Additionally, possession and use of byproduct, source, and special nuclear material at KPS during decommissioning activities is covered by License Condition 2.B, which will remain in effect. Therefore, License Condition 1.I is not needed.

This paragraph will read as follows.

I. Deleted; and

License Condition 1.J

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

This section is proposed for deletion in its entirety. KPS has permanently ceased operation prior to the period of extended operation. Since 10 CFR 50.82(a)(2) prohibits operation of the KPS reactor once the certifications described therein are submitted, KPS will not operate during the period of extended operation. Decommissioning of KPS is not dependent on the requirements of Part 54 for a renewed license. Therefore, requirements that are unique to a renewed license are not needed

This paragraph will read as follows.

J. Deleted.

License Condition 2.B.(1)

2.B.(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess~~, use, and operate~~ the facility at the designated location in Kewaunee County, Wisconsin in accordance with the procedures and limitations set forth in this renewed license;

The language regarding use and operation is proposed for deletion. The license no longer authorizes use and operation of the facility.

This paragraph will read as follows.

2.B.(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess the facility at the designated location in Kewaunee County, Wisconsin in accordance with the procedures and limitations set forth in this renewed license;

License Condition 2.B.(2)

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

The language regarding receipt and use of special nuclear material as reactor fuel is proposed for deletion (and referring to use of reactor fuel in the past tense). The license no longer authorizes use and operation of the facility and only authorizes possession of the existing fuel.

This paragraph will read as follows.

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

License Condition 2.B.(3)

2.B.(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source 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; and possess any byproduct, source and special nuclear material as sealed neutron sources that was used for reactor startup;

The language regarding receipt and use of sealed neutron sources for reactor startup is proposed for deletion. This license condition is revised to reflect authorization only for continued possession of those sources used for reactor startups. The license no longer authorizes use and operation of the facility and this condition will no longer authorize receipt and use of sources used for reactor startup.

This paragraph will read as follows.

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

License Condition 2.C.(1)

2.C.(1) ~~Maximum Power Level~~

~~The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 1772 megawatts (thermal).~~

This section is proposed for deletion in its entirety. KPS has permanently ceased power operation. 10 CFR 50.82(a)(2) prohibits operation of the KPS reactor since the certifications described therein have been submitted.

This paragraph will read as follows.

2.C.(1) Deleted.

License Condition 2.C.(3)

2.C.(3) ~~Fire Protection~~

~~The licensee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the licensee's Fire Plan, and as referenced in the Updated Safety Analysis Report (USAR), and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981), subject to the following provision:~~

~~The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

This section is proposed for deletion in its entirety. KPS has permanently ceased operation. This condition is no longer needed to assure safety by maintaining the ability to achieve and maintain safe shutdown in the event of a fire.

License Condition 2.C.(3), 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 shutdown of the reactor in the event of a fire, is no longer applicable at KPS. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant 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 shutdown and defueled plant is not needed.

This paragraph will read as follows.

2.C.(3) Deleted.

License Condition 2.C.(4)

2.C.(4) Physical Protection

The licensee 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 to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21 is entitled: "Nuclear Management Company Kewaunee Nuclear Power Plant Physical Security Plan (Revision 0)" submitted by letter dated October 18, 2004, and supplemented by letter dated October 21, 2004, July 26, 2005, and May 15, 2006.

~~The licensee shall fully implement and maintain in effect all provisions of the Commission-approved Kewaunee, Millstone, North Anna, and Surry Power Stations Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The CSP was approved by License Amendment No. 210.~~

The second paragraph of this license condition, which addresses the Cyber Security Plan, is proposed for deletion in its entirety. KPS has permanently ceased operation. Therefore, the Cyber Security Plan license condition is no longer required.

This license condition was added to comply with 10 CFR 73.54, which states: "By November 23, 2009 each licensee currently licensed to operate a nuclear power plant under part 50 of this chapter shall submit, as specified in §50.4 and §50.90 of this chapter, a cyber security plan that satisfies the requirements of this section for Commission review and approval." (emphasis added)

However, since the certifications required under 10 CFR 50.82(a) have been submitted, and DEK is no longer authorized to operate a nuclear power plant, 10 CFR 73.54 no longer applies to DEK. Therefore, the Cyber Security Plan required by the second paragraph of this license condition, is no longer required.

The first paragraph of License Condition 2.C.(4) remains unchanged.

License Condition 2.C.(6)

2.C.(6) ~~Steam Generator Upper Lateral Supports~~

~~The design of the steam generator upper lateral supports may be modified by reducing the number of snubbers from four (4) to one (1) per steam generator.~~

This section is proposed for deletion in its entirety. KPS has permanently ceased operation; therefore, the steam generator upper lateral supports are no longer required. 10 CFR 50.82(a)(2) prohibits operation of the KPS reactor since the certifications described therein have been submitted. Therefore, the steam generators will not be operated again in the future. Decommissioning of KPS is not dependent on the requirements for steam generator upper lateral supports.

This paragraph will read as follows.

2.C.(6) Deleted.

License Condition 2.C.(10)

2.C.(10) ~~Mitigation Strategy License Condition~~

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

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

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

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

This section is proposed for deletion in its entirety. KPS has permanently ceased operation; therefore, the mitigation strategy license condition is no longer required.

The NRC issued this license condition on August 2, 2007, to incorporate the requirements for the Interim Compensatory Measures (ICM) Order EA-02-026, Section B.5.b mitigation strategies (dated February 25, 2002). Subsequently, 10 CFR 50.54(hh)(2) became effective on May 26, 2009. This section provides mitigation strategies and response procedure requirements for loss of large areas of the plant due to explosions or fire. However, as stated in 10 CFR 50.54(hh)(3), this section does not apply to a defueled reactor that has submitted the certification for permanent removal of fuel under 10 CFR 50.82(a).

On November 28, 2011, the NRC issued a letter to rescinded Item B.5.b of the ICM Order EA-02-26. Therefore, neither the ICM Order nor 10 CFR 50.54(hh) continue to apply to KPS.

This paragraph will read as follows.

2.C.(10) Deleted.

License Condition 2.C.(14)

2.C.(14) ~~Deferral of Certain Technical Specification Requirements~~

~~Following implementation of License Amendment No. 207, the requirement for the reactor coolant system (RCS) Hot Leg A Temperature Indication to be OPERABLE as required by technical specification (TS) 3.3.3 and TS 3.3.4 may be deferred until startup after the first outage of sufficient duration to repair the RCS Hot Leg A Temperature Indication. Specifically, TS Table 3.3.3-1, Function 3 will only require 1 channel to be OPERABLE, and TS Table B 3.3.4-1, Function 4.a will not be applicable. Following the startup after the first outage of sufficient duration to repair the RCS Hot Leg A Temperature Indication, TS Table 3.3.3-1 Function 3 and TS Table B 3.3.4-1, Function 4.a requirements will be applicable.~~

This section is proposed for deletion in its entirety. KPS has permanently ceased operation; therefore, the reactor coolant system (RCS) hot leg A temperature indication is no longer required. 10 CFR 50.82(a)(2) prohibits operation of the KPS reactor since the certifications described therein have been submitted. Therefore, KPS will not operate the RCS and the hot leg temperature indication is not needed when the plant is in a shutdown and defueled condition. Decommissioning of KPS is not dependent on the requirements for RCS temperature indication.

Additionally, License Condition 2.C.(14) was a one-time allowance which deferred Operability requirements for the RCS hot leg A temperature indication until it was able to be repaired. This condition had previously been satisfied and was no longer needed.

This paragraph will read as follows.

2.C.(14) Deleted.

License Condition 2.C.(15)

2.C.(15) ~~License Renewal License Conditions~~

~~(a) The USAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the USAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, the licensee may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.~~

~~(b) The USAR supplement, as revised, describes certain future activities to be completed prior to and/or during the period of extended operation. The licensee shall complete these activities in accordance with Appendix A of NUREG-1958, "Safety Evaluation Report Related to the Kewaunee Power Station," dated January 2011. The licensee shall notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection.~~

~~(c) All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC.~~

Deletion of License Condition 2.C.(15) was separately requested in License Amendment Request LAR 255 (Reference 8) on April 16, 2013. For completeness, the proposed deletion is reproduced herein in its entirety and with no alteration to the deletion as proposed in LAR 255. Approval of LAR 255 continues to be requested prior to December 22, 2013, as discussed in that submittal. Upon approval of LAR 255, the corresponding change to License Condition 2.C.(15) requested in this amendment request (LAR 256) will have been approved.

License Condition 2.C.(15) was issued concurrent with the Renewed Facility Operating License on February 24, 2011. This license condition (consisting of three parts) is described in Section 1.7, "Summary of Proposed License Conditions," of the Safety Evaluation Report (SER) for KPS dated November 4, 2010 (Reference 4). The KPS license renewal SER was subsequently published as NUREG-1958, "Safety Evaluation Report Related to the License Renewal of the Kewaunee Power Station," issued January 2011.

License Condition 2.C.(15) is proposed for deletion in its entirety. The period of extended operation for KPS begins on December 22, 2013. KPS has permanently ceased operation and has permanently defueled the reactor vessel prior to the start of the period of extended operation. Upon docketing of the certifications of permanent shutdown and permanent removal of fuel from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the KPS reactor, or emplacement or retention of fuel into the reactor vessel. Therefore, KPS will not operate during the period of extended operation and none of the activities that are unique to the renewed license are needed. Decommissioning of KPS is not dependent on the requirements of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," for a renewed license.

License Condition 2.C.(15)(a) is a one-time requirement to update the USAR to include the USAR supplement required by 10 CFR 54.21(d) in the next USAR update as required by 10 CFR 50.71(e). This is duplicative of the existing requirements in 10 CFR 54.21(d) and 10 CFR 50.71(e)(4). Since the USAR update required by this License Condition has been previously completed, this license condition has been satisfied and is therefore no longer needed. License Condition 2.C.(15)(a) also states that DEK may make changes to the programs and activities described in the supplement without prior NRC approval provided that the changes are made pursuant to 10 CFR 50.59 requirements. The requirements of 10 CFR 50.59 will continue to apply to such changes after the license condition is deleted. Therefore, after deletion of this license condition, changes to these programs and activities may be made without prior Commission approval provided that DEK evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59.

License Condition 2.C.(15)(b) pertains to completion of activities described in Appendix A of NUREG-1958, "Safety Evaluation Report Related to the Kewaunee Power Station," dated January 2011. The license condition states that DEK shall notify NRC when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection. Since KPS has permanently ceased operation and defueled the reactor prior to the period of extended operation, these activities are no longer needed. DEK is not aware of the existence of such a license condition in the license of any other U.S. nuclear facility that has permanently ceased operation in accordance with 10 CFR 50.82. The provisions of 10 CFR 54.35, "Requirements during term of renewed license," ensure that all applicable Commission regulations remain in force during the extended period of the renewed license. These regulations provide sufficient requirements for ensuring the safe storage and management of irradiated fuel at a permanently defueled facility. After deletion of this license condition, changes to these activities may be made without prior Commission approval provided that DEK evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59.

The capsules referred to in License Condition 2.C.(15)(c) are not required to confirm continued safe conditions for a permanently defueled facility. As such, this License Condition is also no longer needed.

Upon expiration of the renewed facility operating license on December 21, 2033, activities required for the safe storage and management of irradiated fuel and decommissioning of KPS will continue to be governed in accordance with 10 CFR 50.51, "Continuation of license." 10 CFR 50.51 directs that 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.

Safe storage and management of irradiated fuel at other U.S. nuclear facilities licensed under 10 CFR 50, that have permanently ceased operation in accordance with 10 CFR 50.82, does not rely on the programs and activities associated with 10 CFR 54. As such, the requirements specified in License Condition 2.C.(15) are not needed for assuring safe onsite storage of irradiated fuel at KPS and may be deleted.

This paragraph will read as follows.

2.C.(15) Deleted.

License Condition 2.E

E. ~~This renewed operating license is effective as of the date of issuance and shall expire at midnight on December 21, 2033.~~

This section is proposed for deletion in its entirety. KPS has permanently ceased operation prior to the period of extended operation. 10 CFR 50.82(a)(2) prohibits operation of the KPS reactor since the certifications described therein have been docketed. Therefore, KPS will not operate during the period of extended operation. This License Condition is being replaced by new License Condition 3, which comports to 10 CFR 50.51 in that the possession-only license authorizes ownership and possession of KPS until the Commission notifies the licensee in writing that the license is terminated.

This paragraph will read as follows.

E. Deleted.

License Condition 3

A new License Condition 3 is being proposed to address the permanently defueled possession-only status of the facility and replace existing License Condition 2.E so as to comport to 10 CFR 50.51 in that the possession-only license authorizes ownership and possession of KPS until the Commission notifies the licensee in writing that the license is terminated.

This new License Condition will read as follows.

3. On February 25, 2013, Dominion Energy Kewaunee (DEK) certified that operations at Kewaunee Power Station would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On May 14, 2013, DEK certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

   This license is effective as of the date of issuance and authorizes ownership and possession of Kewaunee Power Station until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

   A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

   B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

3.0 TECHNICAL ANALYSIS

and

4.0 SUMMARY

The following portion of this license amendment request contains the technical analysis for justifying the proposed change, and a summary of the change, for Technical Specifications (TS) Sections 1.0, 2.0, 3.0 through 3.9, 4.0 and 5.0.

A combined chapter, containing a separate description, the proposed change, technical analysis, and summary of the change is provided separately for each TS section. These individual chapters combine to constitute Parts 3.0 and 4.0 of this license amendment request.

For grouping purposes, each separate description, proposed change, technical analysis, and summary of the change for each TS section is labeled as follows.

Txx.1 Description
Txx.2 Proposed Change
Txx.3 Technical Analysis
Txx.4 Summary

The "xx" is a numerical designator that corresponds to the associated TS section (e.g., T3.1 would correspond to the "Description" for TS 3, whereas T37.3. would correspond to the "Technical Analysis" for TS 3.7).

The "**General Analysis Applicable to Proposed Change**" documented in Section 2.0, "Proposed Change," above, is also applicable to the following proposed TS changes.

► TS SECTION 1.0, USE AND APPLICATION ◄

T1.1 DESCRIPTION

The existing TS Section 1.0, "Use and Application," contains the rules of usage for the TS. This section is divided into the following four subsections.

1.1 Definitions – Defines terms used and applicable throughout the TS and Bases.

1.2 Logical Connectors – An explanation of the logical connectors used to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies.

1.3 Completion Times – Establishes the Completion Time convention and provides guidance for its use.

1.4 Frequency – Defines the proper use and application of Frequency requirements.

Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the aspects of the this section that do not apply in a defueled condition are being proposed for deletion.

T1.2 PROPOSED CHANGE

TS Section 1.0, USE AND APPLICATION

All TS in Section 1.0 are being retained, as identified in the table below. Proposed revisions to these TS are as further described below and shown in Attachment 2.

TS Being Deleted	TS Being Revised
1.0 USE AND APPLICATION	
	1.1 Definitions
	1.2 Logical Connectors
	1.3 Completion Times
	1.4 Frequency

There are no corresponding TS Bases sections associated with this TS section.

T1.3 TECHNICAL ANALYSIS

TS Section 1.1, Definitions

TS 1.1, "Definitions," provides defined terms that are applicable throughout the TS and TS Bases. The following definitions are being proposed for deletion because they have no relevance to and therefore no longer apply to the permanently defueled plant status.

Definitions Being Deleted

Term	Definition Being Deleted (summarized)
Axial Flux Difference (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
Core Operating Limits Report (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle.
Dose Equivalent I-131	Dose Equivalent I-131 shall be that concentration of I-131 that alone would produce the same dose when inhaled as the combined activities of specified iodine isotopes actually present. The determination shall be performed using ICRP-30, 1979, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
Dose Equivalent Xe-133	Dose Equivalent Xe-133 shall be that concentration of Xe-133 that alone would produce the same acute dose to the whole body as the combined activities of specified noble gas nuclides actually present. The determination shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
Leakage	LEAKAGE (from the reactor coolant system) shall be (*as summarized*): a. Identified LEAKAGE (that is captured and conducted to collection systems or a sump or collecting tank, into the containment atmosphere, primary to secondary) b. Unidentified LEAKAGE c. Pressure Boundary LEAKAGE
Mode	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

Term	Definition Being Deleted (summarized)
Physics Tests	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.
Quadrant Power Tilt Ratio (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
Rated Thermal Power (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1772 MWt.
Shutdown Margin (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming the specified conditions.
Thermal Power	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

In conjunction with deletion of the term “Mode,” TS Table 1.1-1, “Modes,” is also being deleted.

Definition Being Added

The following definition is being added.

Term	Definition Being Added
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.

The definition of the term Certified Fuel Handler is being added to ensure consistent understanding and application. Further discussion regarding Certified Fuel Handlers is included in the Administrative Controls section of the proposed TS.

TS Section 1.2, Logical 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 the TS. This section continues to apply and is being retained with no changes.

TS Section 1.3, Completion Times

TS 1.3, "Completion Times," establishes the Completion Time convention throughout the TS and provides guidance for its use. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, this section is being revised to comport to the permanently defueled condition.

Statements referring to "operation of the unit" are replaced with "management of irradiated fuel," since operation of the unit is no longer permitted and safe management (storage and movement) of irradiated fuel is the primary objective of the permanently defueled TS.

References to the term "unit" are replaced with the term "facility", because the term unit generally refers to the reactor, which can no longer be operated, whereas the term facility refers to the overall site, including the fuel storage facility.

References to the term "Mode" and "Thermal Power" are deleted, as these terms are no longer applicable to a permanently defueled facility. This includes revisions to examples that include these terms, and replacing them in the examples with generic activities that continue to be applicable in a permanently defueled condition.

A portion of the explanation for Example 1.3-2 is deleted because it pertains to activities that no longer pertain to a permanently defueled condition.

TS Section 1.4, Frequency

TS 1.4, "Frequency," defines the proper use and application of Frequency requirements throughout the TS. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, this section is being revised to comport to the permanently defueled condition.

References to the terms "Mode" and "reactor power" are either deleted or replaced with terms such as "specified condition." These two former terms ("Mode" and "reactor power") are no longer applicable to a permanently defueled facility.

The final paragraph of the TS 1.4 Description section, regarding Notes that modify the Frequency of performance of some surveillances and the applicability of entry restrictions of SR 3.0.4, is being deleted in its entirety. The subsequent subparagraphs regarding any of three conditions being satisfied, which are prefaced as a, b, and c, are also being deleted. None of the surveillances in the proposed TS contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. Therefore, this paragraph is not applicable to the proposed TS LCOs or Surveillance Requirements and may be deleted.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 are deleted because these examples are not needed in a permanently defueled condition. These examples used references to an operating reactor and there are no longer any conditions that pertain to them. The remaining examples are sufficient to explain application of TS frequency requirements.

T1.4 SUMMARY

Current Operating Licensed Condition

TS Section 1.0, "Use and Application," does not contain applicability requirements. As such, all parts of this section can be conservatively defined as being applicable at all times.

Permanently Defueled Condition

TS Section 1.0 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

Conclusion

Retaining TS Section 1, as revised, provides appropriate control over use and application of the KPS TS.

► TS SECTION 2.0, SAFETY LIMITS ◄

T2.1 DESCRIPTION

The existing TS Section 2.0, "Safety Limits," contains limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity from the reactor core and the reactor coolant system. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, these safety limits do not apply and are being proposed for deletion.

T2.2 PROPOSED CHANGE

TS Section 2.0, SAFETY LIMITS (SLs)

All TS in Section 2.0 are being proposed for deletion, as identified in the table below.

TS Being Deleted	TS Being Revised
2.0 SAFETY LIMITS (SLs)	
2.1 Safety Limits (SLs)	
2.2 SL Violations	

The corresponding TS Bases sections are also being deleted to reflect this change.

T2.3 TECHNICAL ANALYSIS

Section 2.0 TS That Are Not Applicable When Defueled

TS 2.1 and 2.2 do not currently apply with the reactor defueled.

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 (RCS) SL.

The restrictions of the SL promulgated in TS 2.1.1 prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. TS 2.1.1 is applicable in Modes 1 and 2.

TS 2.1.2 promulgates requirements on parameters to protect the integrity of the reactor coolant system (RCS) against overpressure. TS 2.1.2 is applicable in Modes 1, 2, 3, 4, and 5.

TS 2.2, "Safety Limit 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) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific process variables, are no longer applicable. Based on the above, the proposed deletion of TS related to these safety limits is acceptable.

T2.4 SUMMARY

Current Operating Licensed Condition

TS Section 2.0 does not currently apply with the reactor defueled.

Permanently Defueled Condition

Since TS Section 2.0 does not apply with the reactor defueled, the individual TS contained therein are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 2.0 is acceptable.

► TS SECTION 3.0, LIMITING CONDITION FOR OPERATION APPLICABILITY ◄ SURVEILLANCE REQUIREMENT APPLICABILITY

T3.1 DESCRIPTION

The existing TS Section 3.0, "Limiting Condition for Operation (LCO) Applicability," and "Surveillance Requirement (SR) Applicability," contains general requirements applicable to all Specifications. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the general requirements that do not apply in a defueled condition are being proposed for deletion.

T3.2 PROPOSED CHANGE

TS Section 3.0, LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY; SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

Both TS in Section 3.0 are being retained, as identified in the table below. Certain specific LCOs within TS 3.0 are being proposed for deletion. Proposed revisions to these TS (including the LCOs being deleted) are as further described below and shown in Attachment 2.

TS Being Deleted	TS Being Revised
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	
	3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
	3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

The corresponding TS Bases sections are also being revised to reflect this change.

T3.3 TECHNICAL ANALYSIS

TS 3.0, LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

TS 3.0, "Limiting Condition for Operation (LCO) Applicability," consists of LCO 3.0.1 through LCO 3.0.8. LCO 3.0.1 through LCO 3.0.8 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 is in the Modes or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.1 is being retained in the permanently defueled TS with the proposed revisions shown in Attachment 2.

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. Additionally, references to LCO 3.0.7 and LCO 3.0.8 are also being deleted to conform to deletion of those two LCOs 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 being retained in the permanently defueled TS unchanged.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met. LCO 3.0.3 is only applicable in Modes 1, 2, 3, and 4. LCO 3.0.3 is the only LCO in this section that does not currently 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 is no longer 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 being retained in the permanently defueled TS with the proposed revisions shown in Attachment 2.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, references to operating modes are no longer relevant and are therefore being deleted. Additionally, a discussion pertaining to shutdown of the unit is likewise being deleted.

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 being retained in the permanently defueled TS unchanged.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the TS. This exception is justified because the actions that are required to ensure the facility is maintained in a safe condition are specified in the support system LCO's Required Actions. When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions.

LCO 3.0.6 is being retained in the permanently defueled TS with the proposed revisions shown in Attachment 2.

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 Safety Function Determination Program (SFDP), contained in TS 5.5.13, "Safety Function Determination Program (SFDP)," which directs cross train checks of multiple and redundant safety systems, no longer apply. Therefore, reference to the SFDP is being deleted to conform to deletion of TS 5.5.13 as described in Section T5.3 of this document.

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 associated with Test Exception LCO 3.1.8, which is being deleted as described in Section T31.3 of this document.

LCO 3.0.7 is being proposed for deletion in its entirety.

Because TS LCO 3.1.8 is being deleted, LCO 3.0.7 is no longer germane. Therefore, the proposed deletion of LCO 3.0.7 is acceptable.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s).

LCO 3.0.8 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, all systems associated with snubbers are no longer required to be operable. As such, the allowance provided by LCO 3.0.8 is no longer needed. Therefore, the proposed deletion of LCO 3.0.8 is acceptable.

TS 3.0, SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

TS 3.0, "Surveillance Requirement (SR) Applicability," consists of SR 3.0.1 through SR 3.0.4. 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.

SR 3.0.1 is being retained in the permanently defueled TS with the proposed revisions shown in Attachment 2.

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 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 is being retained in the permanently defueled TS unchanged.

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 requirement has not been completed within the specified Frequency.

SR 3.0.3 is being retained in the permanently defueled TS unchanged.

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.

SR 3.0.4 is being retained in the permanently defueled TS with the proposed revisions shown in Attachment 2.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, references to operating modes are no longer relevant and are therefore being deleted. Additionally, a discussion pertaining to shutdown of the unit is likewise being deleted.

T3.4 <u>SUMMARY</u>

Current Operating Licensed Condition

TS Section 3.0, "Limiting Condition for Operation (LCO) Applicability," and "Surveillance Requirement (SR) Applicability," does not contain applicability requirements (except for LCO 3.0.3). As such, all parts of this section (except for LCO 3.0.3) can be conservatively defined as being applicable at all times.

LCO 3.0.3 does not currently apply with the reactor defueled.

Permanently Defueled Condition

Since LCO 3.0.3 does not apply with the reactor defueled, it is not needed for a permanently defueled condition. As such, it may be deleted with no impact on continued safe operation of the facility.

LCO 3.0.7 and LCO 3.0.8 pertain only to TS that are being deleted and systems to which TS no longer apply. As such, they may be deleted with no impact on continued safe operation of the facility.

LCO 3.0.1, LCO 3.0.2, LCO 3.0.4, LCO 3.0.5, LCO 3.0.6, SR 3.0.1, SR 3.0.2, SR 3.0.3, and SR 3.0.4 in Section 3.0 will remain applicable with the reactor permanently defueled. As such, they are being retained and revised to reflect a permanently defueled condition.

Conclusion

Deleting LCO 3.0.3, LCO 3.0.7, and LCO 3.0.8 in Section 3.0 is acceptable.

Retaining LCO 3.0.1, LCO 3.0.2, LCO 3.0.4, LCO 3.0.5, LCO 3.0.6, SR 3.0.1, SR 3.0.2, SR 3.0.3, and SR 3.0.4 in Section 3.0, as proposed, ensures appropriate requirements for application of LCOs and SRs.

► TS SECTION 3.1, REACTIVITY CONTROL SYSTEMS ◄

T31.1 DESCRIPTION

The existing TS Section 3.1, "Reactivity Control Systems," contains Limiting Conditions for Operation (LCOs) that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of a fission product barrier. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, these LCOs (and associated Surveillance Requirements (SRs)) do not apply in a defueled condition and are being proposed for deletion.

T31.2 PROPOSED CHANGE

TS Section 3.1, Reactivity Control Systems

All TS in Section 3.1 are being proposed for deletion, as identified in the table below.

TS Being Deleted	TS Being Revised
3.1 REACTIVITY CONTROL SYSTEMS	
3.1.1 SHUTDOWN MARGIN (SDM)	
3.1.2 Core Reactivity	
3.1.3 Moderator Temperature Coefficient (MTC)	
3.1.4 Rod Group Alignment Limits	
3.1.5 Shutdown Bank Insertion Limits	
3.1.6 Control Bank Insertion Limits	
3.1.7 Rod Position Indication	
3.1.8 PHYSICS TESTS Exceptions - MODE 2	

The corresponding TS Bases sections are also being deleted to reflect this change.

T31.3 TECHNICAL ANALYSIS

Section 3.1 TS That Are Not Applicable When Defueled

None of the TS in Section 3.1 currently apply with the reactor defueled.

TS 3.1.1, "SHUTDOWN MARGIN (SDM)," specifies requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). TS 3.1.1 is applicable in Mode 2 with $k_{eff} < 1.0$ and in Modes 3, 4, and 5.

TS 3.1.2, "Core Reactivity," specifies requirements for two independent reactivity control systems. The reactivity control systems provided shall be capable of making and holding the core sub-critical from any hot standby or hot operation condition. TS 3.1.2 is applicable in Modes 1 and 2.

TS 3.1.3, "Moderator Temperature Coefficient (MTC)," specifies requirements to ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. The MTC relates a change in core reactivity to a change in reactor coolant temperature. TS 3.1.2 is applicable in Mode 1 and Mode 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, and in Modes 1, 2, and 3 for the lower MTC limit.

TS 3.1.4, "Rod Group Alignment Limits," specifies requirements for limits on shutdown or control rod alignments, to ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod Operability ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. Operability requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the rod cluster control assemblies and banks maintain the correct power distribution and rod alignment. TS 3.1.4 is applicable in Modes 1 and 2.

TS 3.1.5, "Shutdown Bank Insertion Limits," specifies limits on shutdown bank physical insertion to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required shutdown margin (SDM) following a reactor trip. TS 3.1.5 is applicable in Modes 1 and 2.

TS 3.1.6, "Control Bank Insertion Limits," specifies limits on control bank sequence, overlap, and physical insertion for the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion. TS 3.1.6 is applicable in Mode 1 and in Mode 2 with $k_{eff} \geq 1.0$.

TS 3.1.7, "Rod Position Indication," specifies requirements for OPERABILITY of the rod position indicators to determine rod positions and thereby ensure compliance with the rod alignment and insertion limits. TS 3.1.7 is applicable in Modes 1 and 2.

TS 3.1.8, "Physics Tests Exceptions – Mode 2," permits relaxation of existing TS Limiting Conditions for Operation (LCOs) to allow certain PHYSICS TESTS to be performed. TS 3.1.8 is applicable during performance of Physics Tests initiated in Mode 2.

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address their associated specific plant equipment, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.1 is acceptable.

T31.4 SUMMARY

Current Operating Licensed Condition

None of these TS currently apply with the reactor defueled.

Permanently Defueled Condition

Since these TS do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.1 is acceptable.

► TS SECTION 3.2, POWER DISTRIBUTION LIMITS ◄

T32.1 DESCRIPTION

The existing TS Section 3.2, "Power Distribution Limits," contains Limiting Conditions for Operation (LCOs) that provide for appropriate control of process variables, design features, or operating restrictions that are required to protect the integrity of a fission product barrier. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion.

T32.2 PROPOSED CHANGE

TS Section 3.2, Power Distribution Limits

All TS in Section 3.2 are being proposed for deletion, as identified in the table below.

TS Being Deleted		TS Being Revised
3.2	**POWER DISTRIBUTION LIMITS**	
3.2.1	Heat Flux Hot Channel Factor (FQ(Z))	
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	
3.2.3	AXIAL FLUX DIFFERENCE (AFD)	
3.2.4	QUADRANT POWER TILT RATIO (QPTR)	

The corresponding TS Bases sections are also being deleted to reflect this change.

T32.3 TECHNICAL ANALYSIS

Section 3.2 TS That Are Not Applicable When Defueled

None of the TS in Section 3.2 currently apply with the reactor defueled.

TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," specifies limits on the values of $F_Q(Z)$ to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core. TS 3.2.1 is applicable in Mode 1.

TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," specifies limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses. $F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. TS 3.2.2 is applicable in Mode 1.

TS 3.2.3, "Axial Flux Difference (AFD)," specifies limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control. TS 3.2.3 is applicable in Mode 1 greater than or equal to 50% Rated Thermal Power.

TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," specifies limits on the values of QPTR in order to ensure that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. The power density at any point in the core must be limited so that the fuel design criteria are maintained. TS 3.2.4 is applicable in Mode 1 with thermal power greater 50% Rated Thermal Power.

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Mode 1. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address their associated specific plant equipment, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.2 is acceptable.

T32.4 SUMMARY

Current Operating Licensed Condition

None of these TS currently apply with the reactor defueled.

Permanently Defueled Condition

Since these TS do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.2 is acceptable.

► TS SECTION 3.3, INSTRUMENTATION ◄

T33.1 DESCRIPTION

The existing TS Section 3.3, "Instrumentation," contains Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of sensing and control instrumentation required for safe operation of the facility, including the plant being in a defueled condition. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition, or are not needed for accident mitigation in the defueled condition, are being proposed for deletion.

T33.2 PROPOSED CHANGE

TS Section 3.3, Instrumentation

All TS in Section 3.3 are being proposed for deletion, as identified in the table below.

TS Being Deleted		TS Being Revised
3.3	**INSTRUMENTATION**	
3.3.1	Reactor Protection System (RPS) Instrumentation	
3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	
3.3.3	Post Accident Monitoring (PAM) Instrumentation	
3.3.4	Dedicated Shutdown System	
3.3.5	Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation	
3.3.6	Containment Purge and Vent Isolation Instrumentation	
3.3.7	Control Room Post Accident Recirculation (CRPAR) System Actuation Instrumentation	

The corresponding TS Bases sections are also being deleted to reflect this change.

T33.3 TECHNICAL ANALYSIS

Section 3.3 TS That Are Not Applicable When Defueled

TS 3.3.1, 3.3.2, 3.3.3, and 3.3.4 do not currently apply with the reactor defueled.

TS 3.3.1, "Reactor Protection System (RPS) Instrumentation," specifies requirements for the RPS instrumentation system to maintain the safety limits during all anticipated operational occurrences and mitigates the consequences of design basis accidents in all Modes in which the rod control system is capable of rod withdrawal or one or more rods are not fully inserted. The RPS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and reactor coolant system (RCS) pressure boundary during anticipated operational occurrences and to assist the engineered safety features (ESF) systems in mitigating accidents. The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings in terms of parameters directly monitored by the RPS, as well as specifying LCOs on other reactor system parameters and equipment performance. TS 3.3.1 is applicable in Modes 1, 2, 3, 4, and 5 (according to specific applicability requirements for each RPS function listed in TS Table 3.3.1-1).

TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," specifies requirements for the ESFAS instrumentation system to ensure that ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS. Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 50.67 limits. TS LCO 3.3.2 requires all instrumentation performing an ESFAS Function, listed in TS Table 3.3.2-1, to be Operable. TS 3.3.2 is applicable in Modes 1, 2, 3, and 4 (according to specific applicability requirements for each ESFAS function listed in TS Table 3.3.2-1).

TS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," provides Operability requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this TS LCO addresses Regulatory Guide 1.97 instruments that have been designated Category 1, non-Type A. The Operability of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. The specific instrument Functions are listed in TS Table 3.3.3-1. The PAM Instrumentation LCO is applicable in Modes 1, 2, and 3.

TS 3.3.4, "Dedicated Shutdown System," provides the Operability requirements for the instrumentation and controls necessary to place and maintain the unit in Mode 3 from a location other than the control room. The instrumentation and controls required are listed in TS Bases Table B 3.3.4-1. The Dedicated Shutdown System LCO is applicable in Modes 1, 2, and 3.

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of TS 3.3.1, 3.3.2, 3.3.3, and 3.3.4 is acceptable.

TS 3.3.5, Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation

TS 3.3.5, "Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation," specifies that one channel per bus for the Safeguards Bus Undervoltage (loss of voltage) Function and one channel per bus for the Safeguards Bus Second Level Undervoltage (degraded voltage) Function be Operable. The LOOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function during any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS). The required channels of LOOP DG start instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in the USAR in which a loss of offsite power is assumed. The LOOP DG start instrumentation supports EDG Operability. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

TS 3.3.5 is applicable in Modes 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these Modes. TS 3.3.5 is also applicable whenever the associated DG is required to be Operable by LCO 3.8.2, "AC Sources – Shutdown," to ensure that the automatic start of the DG is available when needed.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.3.5 is no longer needed for assuring the appropriate functional capability of the LOOP DG start instrumentation for safe operation of the facility when the reactor is in Modes 1, 2, 3, or 4.

As discussed in the justification for deleting TS 3.8.2 below, the requirement for EDGs is being deleted from the TS because, after the plant is permanently shutdown and defueled, there are no design basis events that rely on the EDGs for mitigation. Since TS 3.3.5 exists solely to support EDG Operability, the elimination of the need for EDGs also obviates the need for their support systems. Since LOOP DG start instrumentation is no longer needed, TS 3.3.5 may be deleted.

Based on the above, the proposed deletion of TS 3.3.5 for LOOP DG start instrumentation is acceptable.

TS 3.3.6, Containment Purge and Vent Isolation Instrumentation

TS 3.3.6, "Containment Purge and Vent Isolation Instrumentation," specifies requirements for instrumentation designed to close the containment isolation valves in the containment vessel air handling system, consisting of the containment air cooling and containment purge and vent systems. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident.

TS 3.3.6 is applicable in Modes 1, 2, 3, and 4. Certain functions are also applicable during movement of irradiated fuel assemblies within containment (according to specific applicability requirements for each function listed in TS Table 3.3.6-1).

TS 3.3.6 is related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 4 and also during movement of irradiated fuel assemblies within containment. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, which thereby precludes entry into Modes 1 through 4. The prohibition on placing fuel in the reactor vessel also precludes movement of irradiated fuel within containment. Therefore, TS 3.3.6, which only address specific plant systems, control of process variables, design features, or operating restrictions associated with the containment is no longer applicable and may be deleted.

Based on the above, the proposed deletion of TS 3.3.6 is acceptable.

TS 3.3.7, Control Room Post Accident Recirculation (CRPAR) System Actuation Instrumentation

TS 3.3.7, "Control Room Post Accident Recirculation (CRPAR) System Actuation Instrumentation," specifies requirements to ensure that instrumentation necessary to initiate the CRPAR System is Operable. The LCO requires two trains of actuation logic and relays, one control room vent radiation monitor (R-23), and actuation capability upon safety injection (SI).

The actuation instrumentation consists of a single radiation monitor (R-23) located on the common discharge of the outlet of the control room air conditioning fan units. A high radiation signal from the detector will initiate both trains of the CRPAR System. The control room operator can also start the CRPAR fan(s) by manual switches in the control room. The CRPAR System is also actuated by a safety injection (SI) signal.

The CRPAR System acts to terminate the normal supply of unfiltered outside air to the control room, both CRPAR fans are started, the flow path through the Emergency Filtration System is opened, and a portion of the return air volume is filtered to remove airborne contaminants and airborne radioactivity, then mixed with the recirculated return air. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post-accident operations by minimizing the radiation exposure of control room personnel.

The radiation monitor actuation of the CRPAR System is a backup for the SI signal actuation. This ensures initiation of the CRPAR System during a loss of coolant accident or steam generator tube rupture when an initiation of SI is anticipated. In addition, the radiation monitor actuation of the CRPAR System is the primary means to ensure control room habitability in the event of a locked rotor accident.

The radiation monitor actuation of the CRPAR System in Modes 5 and 6 and during movement of irradiated fuel assemblies is the primary means to ensure control room habitability in the event of a fuel handling, volume control tank, or waste gas decay tank rupture accident.

TS 3.3.7 is applicable in Modes 1, 2, 3, 4, 5, 6, and during movement of irradiated fuel assemblies. The Applicability for the CRPAR actuation on the ESFAS safety injection functions are specified in LCO 3.3.2 (Modes 1, 2, 3, and 4).

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.7.10 is no longer needed for assuring the appropriate functional capability of the CRPAR system, including the CRPAR system actuation instrumentation, for safe operation of the facility when the reactor is in Modes 1 through 6.

As discussed in the justification for deleting TS 3.7.10, CRPAR System, (in Section T37.3 below), the requirement for the CRPAR System is being deleted from the TS because it is not required for providing airborne radiological protection for the control room operators in the event of a design basis event (fuel handling accident). Since TS 3.3.7 exists solely to support CRPAR system Operability, the elimination of the need for the CRPAR system also obviates the need for its support systems. Since CRPAR System actuation instrumentation is no longer needed, TS 3.3.7 may be deleted.

Based on the above, the proposed deletion of TS 3.3.7 for the CRPAR system actuation instrumentation is acceptable.

T33.4 SUMMARY

Current Operating Licensed Condition

TS 3.3.1, 3.3.2, 3.3.3, and 3.3.4 do not currently apply with the reactor defueled.

TS 3.3.5, 3.3.6, and 3.3.7 remain applicable with the reactor defueled.

Permanently Defueled Condition

Since TS 3.3.1, 3.3.2, 3.3.3, and 3.3.4 do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.3.5, 3.3.6, and 3.3.7 are not needed for accident mitigation in the permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.3 is acceptable.

► TS SECTION 3.4, REACTOR COOLANT SYSTEM (RCS) ◄

T34.1 DESCRIPTION

The existing TS Section 3.4, "Reactor Coolant System (RCS)," contains Limiting Conditions for Operation (LCOs) that provide for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply (or are no longer needed) in a defueled condition are being proposed for deletion.

T34.2 PROPOSED CHANGE

TS Section 3.4, Reactor Coolant System (RCS)

All TS in Section 3.4 are being proposed for deletion, as identified in the table below.

TS Being Deleted		TS Being Revised
3.4	**REACTOR COOLANT SYSTEM (RCS)**	
3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	
3.4.2	RCS Minimum Temperature for Criticality	
3.4.3	RCS Pressure and Temperature (P/T) Limits	
3.4.4	RCS Loops - MODES 1 and 2	
3.4.5	RCS Loops - MODE 3	
3.4.6	RCS Loops - MODE 4	
3.4.7	RCS Loops - MODE 5, Loops Filled	
3.4.8	RCS Loops - MODE 5, Loops Not Filled	
3.4.9	Pressurizer	
3.4.10	Pressurizer Safety Valves	
3.4.11	Pressurizer Power Operated Relief Valves (PORVs)	
3.4.12	Low Temperature Overpressure Protection (LTOP) System	
3.4.13	RCS Operational LEAKAGE	

TS Being Deleted	TS Being Revised
3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	
3.4.15 RCS Leakage Detection Instrumentation	
3.4.16 RCS Specific Activity	
3.4.17 Steam Generator (SG) Tube Integrity	

The corresponding TS Bases sections are also being deleted to reflect this change.

T34.3 TECHNICAL ANALYSIS

Section 3.4 TS That Are Not Applicable When Defueled

None of the TS in Section 3.4, except for TS 3.4.3, currently apply with the reactor defueled.

TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," specifies process variables requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed in the safety analyses. TS 3.4.1 is applicable in Mode 1.

TS 3.4.2, "RCS Minimum Temperature for Criticality," specifies requirements for RCS loop average temperature before the reactor can be made critical and while the reactor is critical. Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the hot zero power temperature, which is assumed in the safety analysis. TS 3.4.2 is applicable in Mode 1 and in Mode 2 with $k_{eff} \geq 1.0$, since the reactor can only be critical ($k_{eff} \geq 1.0$) in these Modes.

TS 3.4.4, "RCS Loops - MODES 1 and 2," specifies requirements to ensure heat removal capability of the RCS loops with the reactor in Modes 1 and 2. The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service. TS 3.4.4 is applicable in Modes 1 and 2.

TS 3.4.5, "RCS Loops - MODE 3," specifies requirements to ensure heat removal capability of the RCS loops with the reactor in Mode 3. In Mode 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the SGs, to the secondary plant fluid. TS 3.4.5 is applicable in Mode 3.

TS 3.4.6, "RCS Loops - MODE 4," specifies requirements to ensure heat removal capability of the RCS loops with the reactor in Mode 4. In Mode 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. TS 3.4.6 is applicable in Mode 4.

TS 3.4.7, "RCS Loops - MODE 5, Loops Filled," specifies requirements to ensure heat removal capability of the RCS loops with the reactor in Mode 5 with the RCS loops filled with coolant. In Mode 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the component cooling water via the RHR heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs, via natural circulation, are specified as a backup means for redundancy. TS 3.4.7 is applicable in Mode 5 with the RCS loops filled.

TS 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," specifies requirements to ensure heat removal capability of the RCS loops with the reactor in Mode 5 with the RCS loops not filled with coolant. In Mode 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the component cooling water via the RHR heat exchangers. The SGs are not available as a heat sink when the loops are not filled. TS 3.4.8 is applicable in Mode 5 with the RCS loops not filled.

TS 3.4.9, "Pressurizer," specifies Operability requirements for the RCS pressurizer. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. In Modes 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. TS 3.4.9 is applicable in Modes 1, 2, and 3.

TS 3.4.10, "Pressurizer Safety Valves," specifies Operability and lift setpoint parameters for the pressurizer safety valves. The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The pressurizer safety valves are designed to prevent the system pressure from exceeding the system Safety Limit, 2735 psig, which is approximately 110% of the design pressure. Overpressure protection is required in Modes 1, 2, 3, 4, and 5. However, in Mode 3 when any RCS cold leg temperature is ≤ 356°F, Modes 4, 5, and Mode 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." TS 3.4.10 is applicable in Modes 1, 2, and 3.

TS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," specifies Operability requirements for the PORVs and their associated block valves. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following an analyzed transient. In addition, the PORVs minimize challenges to the pressurizer safety valves. TS 3.4.11 is applicable in Modes 1, 2, and 3.

TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," specifies requirements for controlling RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. TS LCO 3.4.12 provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. In Modes 1, 2, and in Mode 3 when any RCS cold leg temperature is > 356°F, the pressurizer safety valves will prevent RCS pressure from exceeding limits. At 356°F and below, overpressure prevention falls to the Operable RHR System LTOP overpressure relief valve or to a depressurized RCS and a sufficient sized RCS vent. TS 3.4.12 is applicable in Mode 3 when any indicated RCS cold leg temperature is ≤ 356°F, in Modes 4, 5, and in Mode 6 when the reactor vessel head is on. When the reactor vessel head is off, overpressurization cannot occur.

TS 3.4.13, "RCS Operational LEAKAGE," specifies process variable limits and operating restrictions for RCS pressure boundary leakage, unidentified RCS leakage, identified RCS leakage, and primary to secondary leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event. Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. Other accidents or transients also involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR), locked reactor coolant pump rotor, and control rod ejection. The primary to secondary leakage contaminates the secondary fluid. TS 3.4.13 is applicable in Modes 1, 2, 3, and 4. In Modes 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for leakage.

TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," specifies process variable limits and operating restrictions for RCS PIV leakage and the RHR system interlock function. 10 CFR 50.2, 10 CFR 50.55a(c), and KPS USAR, Section 1.8, Criterion 51, discuss reactor coolant pressure boundary valves, which are normally closed valves in series within the reactor coolant pressure boundary (RCPB) that separate the high pressure RCS from an attached low pressure system. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems (intersystem LOCA). PIVs are provided to isolate the RCS from the Residual Heat Removal (RHR) System. Ensuring the RHR system interlock function that prevents the RHR hot leg suction valves from being opened is Operable ensures that RCS pressure will not pressurize the RHR system beyond its design pressure. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. TS 3.4.14 is applicable in Modes 1, 2, 3, and 4. In Modes 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

TS 3.4.15, "RCS Leakage Detection Instrumentation," specifies Operability requirements for RCS leakage detection instrumentation. Leakage detection systems are provided to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, they provide an early indication or warning signal to permit proper evaluation of RCS leakage into the containment area. TS LCO 3.4.15 requires instruments of diverse monitoring principles to be Operable to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS leakage indicates possible RCPB degradation. TS 3.4.15 is applicable in Modes 1, 2, 3, and 4.

TS 3.4.16, "RCS Specific Activity," specifies process variable limits and operating restrictions for Dose Equivalent I^{131} and Dose Equivalent Xe^{133}. The TS LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate RG 1.183 acceptance criteria following a MSLB or SGTR accident. TS 3.4.16 is applicable in Modes 1, 2, 3, and 4.

TS 3.4.17, "Steam Generator (SG) Tube Integrity," specifies requirements to ensure the reactor coolant pressure boundary integrity function of the SG. The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. TS 3.4.17 is applicable in Modes 1, 2, 3, and 4.

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 6. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which address plant equipment associated with the reactor coolant system, are no longer applicable. Based on the above, the proposed deletion of the above described TS in Section 3.4 is acceptable.

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits

TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, criticality, and data for the maximum rate of change of reactor coolant temperature. Each P/T limit curve defines an acceptable region for normal operation. The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary. TS 3.4.3 is applicable at all times.

The purpose for TS LCO 3.4.3 during normal operation of the RCS is to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary, an unanalyzed condition.

The RCS P/T limits in LCO 3.4.3 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.

10 CFR 50.82, "Termination of license," states that, "upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel..., the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel." 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.

T34.4 <u>SUMMARY</u>

Current Operating Licensed Condition

None of these TS, except TS 3.4.3, currently apply with the reactor defueled.

Permanently Defueled Condition

Since these TS, except TS 3.4.3, do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," applies at all times. However, the system that this specification pertains to (i.e., the RCS) is no longer allowed to be operated. Therefore, TS 3.4.3 is not needed for a permanently defueled condition and may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.4 is acceptable.

► TS SECTION 3.5, EMERGENCY CORE COOLING SYSTEMS (ECCS) ◄

T35.1 DESCRIPTION

The existing TS Section 3.5, "Emergency Core Cooling Systems (ECCS)," contains Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of ECCS equipment required for mitigation of design basis accidents or transients so as to protect the integrity of a fission product barrier. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion.

T35.2 PROPOSED CHANGE

TS Section 3.5, Emergency Core Cooling Systems (ECCS)

All TS in Section 3.5 are being proposed for deletion, as identified in the table below.

TS Being Deleted		TS Being Revised
3.5	**EMERGENCY CORE COOLING SYSTEMS (ECCS)**	
3.5.1	Accumulators	
3.5.2	ECCS - Operating	
3.5.3	ECCS - Shutdown	
3.5.4	Refueling Water Storage Tank (RWST)	

The corresponding TS Bases sections are also being deleted to reflect this change.

T35.3 TECHNICAL ANALYSIS

Section 3.5 TS That Are Not Applicable When Defueled

None of the TS in Section 3.5 currently apply with the reactor defueled.

TS 3.5.1, "Accumulators," specifies requirements for the safety injection accumulators to ensure they are capable of supplying water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide reactor coolant system (RCS) makeup for a small break LOCA. TS 3.5.1 is applicable in Modes 1 and 2, and in Mode 3 with RCS pressure greater than 1000 psig.

TS 3.5.2, "ECCS – Operating," specifies requirements for the emergency core cooling system (ECCS) trains so as to provide core cooling and negative reactivity to ensure that the reactor core is protected after a loss of coolant accident (LOCA), control rod ejection accident, steam line break accident, and steam generator tube rupture (SGTR). The ECCS consists of two separate subsystems: safety injection (SI) (high head) and residual heat removal (RHR) (low head). TS 3.5.2 is applicable in Modes 1, 2, and 3.

TS 3.5.3, "ECCS – Shutdown," specifies requirements for the ECCS trains with the reactor in hot shutdown (Mode 4). In Mode 4, only a single ECCS train is required. The KPS Licensing Basis does not require performance of an analysis to determine the effects of a LOCA occurring in Mode 4, nor does it require an analysis to prove ECCS equipment capability to mitigate a Mode 4 LOCA. Massive failure of Class I piping (e.g., RCS) in Mode 4 is not credible. However, Technical Specifications require certain ECCS subsystems to be Operable in Mode 4 to ensure sufficient ECCS flow is available to the core and adequate core cooling is maintained following a loss of RCS inventory in Mode 4. TS 3.5.3 is only applicable in Mode 4.

TS 3.5.4, "Refueling Water Storage Tank (RWST)," specifies requirements for RWST Operability. During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. TS 3.5.4 is applicable in Modes 1, 2, 3, and 4 because RWST Operability requirements are dictated by ECCS and Containment Spray System Operability requirements. Since both the ECCS and the Containment Spray System must be Operable in Modes 1, 2, 3, and 4, the RWST must also be Operable to support their operation.

The above TS are related to assuring the appropriate functional capability of emergency core cooling systems (ECCS) required for mitigation of design basis accidents only when the reactor is in Modes 1 through 4. However, operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which address the ECCS, are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.5 is acceptable.

T35.4 <u>SUMMARY</u>

Current Operating Licensed Condition

None of these TS currently apply with the reactor defueled.

Permanently Defueled Condition

Since these TS do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.5 is acceptable.

► TS SECTION 3.6, CONTAINMENT SYSTEMS ◄

T36.1 DESCRIPTION

The existing TS Section 3.6, "Containment Systems," contains Limiting Conditions for Operation (LCOs) that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of a fission product barrier; and appropriate functional capability of engineered safety features (ESF) equipment required for mitigation of design basis accidents or transients so as to protect the integrity of a fission product barrier. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion.

T36.2 PROPOSED CHANGE

TS Section 3.6, Containment Systems

All TS in Section 3.6 are being proposed for deletion, as identified in the table below.

TS Being Deleted		TS Being Revised
3.6	CONTAINMENT SYSTEMS	
3.6.1	Containment	
3.6.2	Containment Air Locks	
3.6.3	Containment Isolation Valves	
3.6.4	Containment Pressure	
3.6.5	Containment Air Temperature	
3.6.6	Containment Spray and Cooling Systems	
3.6.7	Spray Additive System	
3.6.8	Shield Building	
3.6.9	Vacuum Relief Valves	
3.6.10	Shield Building Ventilation System (SBVS)	

The corresponding TS Bases sections are also being deleted to reflect this change.

T36.3 TECHNICAL ANALYSIS

Section 3.6 TS That Are Not Applicable When Defueled

None of the TS in Section 3.6 currently apply with the reactor defueled.

TS 3.6.1, "Containment," specifies requirements for the containment to ensure it is capable of withstanding the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate. The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions. TS 3.6.1 is applicable in Modes 1, 2, 3, and 4. In Modes 5 and 6, the probability and consequences of a release are reduced due to the pressure and temperature limitations of these Modes. Therefore, containment is not required to be Operable in Modes 5 or 6 to prevent leakage of radioactive material from containment.

TS 3.6.2, "Containment Air Locks," specifies requirements for the structural integrity and leak tightness of the containment air locks. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each containment air lock's structural integrity and leak tightness is essential to the successful mitigation of such an event. TS 3.6.2 is applicable in Modes 1, 2, 3, and 4, consistent with the applicability requirement of TS 3.6.1, Containment.

TS 3.6.3, "Containment Isolation Valves," specifies requirements for the isolation capability of the containment via the containment isolation valves. Containment isolation valves form a part of the containment boundary and their Operability supports leak tightness of the containment. TS 3.6.3 is applicable in Modes 1, 2, 3, and 4, consistent with the applicability requirement of TS 3.6.1, Containment.

TS 3.6.4, "Containment Pressure," specifies limitations on internal containment pressure. Containment internal pressure is an initial condition used in the design basis accident (DBA) analyses to establish the maximum peak containment internal pressure. Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the maximum allowed containment internal pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent and simultaneous start of all four containment fan-coil units and both trains of the Containment Spray System. TS 3.6.4 is applicable in Modes 1, 2, 3, and 4.

TS 3.6.5, "Containment Air Temperature," specifies limitations on containment average air temperature. Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function. TS 3.6.5 is applicable in Modes 1, 2, 3, and 4.

TS 3.6.6, "Containment Spray and Cooling Systems," specifies Operability requirements for containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. The Containment Spray System consists of two separate trains. Each train includes a containment spray pump and two spray headers. The RWST supplies borated water to the Containment Spray System during the injection phase of operation. Two trains of containment cooling are provided. Each train of two fan-coil units is supplied with cooling water from a separate train of service water. During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment pressure and temperature below the design limits. Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. TS 3.6.6 is applicable in Modes 1, 2, 3, and 4. In Modes 5 and 6, the probability and consequences of a release are reduced due to the pressure and temperature limitations of these Modes. Thus, the Containment Spray System and the Containment Cooling System are not required to be Operable in Modes 5 and 6.

TS 3.6.7, "Spray Additive System," specifies Operability requirements for the Spray Additive System. The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a DBA. This system is necessary to reduce the release of radioactive material to the environment in the event of a DBA. TS 3.6.7 is applicable in Modes 1, 2, 3, and 4.

TS 3.6.8, "Shield Building," specifies Operability requirements for the shield building. The shield building is a concrete structure that surrounds the steel containment vessel. Maintaining shield building Operability prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, TS 3.6.8 is applicable in Modes 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere. In Modes 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these Modes. Therefore, TS 3.6.8 is not applicable in Mode 5 or 6.

TS 3.6.9, "Vacuum Relief Valves," specifies requirements for the vacuum relief valves. The purpose of the vacuum relief lines is to protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). The vacuum relief valves must also perform the containment isolation function in a containment high pressure event. Because the vacuum relief valves support the containment, TS 3.6.9 is applicable in Modes 1, 2, 3, and 4, consistent with the applicability requirement of TS 3.6.1, Containment.

TS 3.6.10, "Shield Building Ventilation System (SBVS)," specifies requirements to ensure that the SBVS reduces the radioactive content in the shield building atmosphere following a design basis accident (DBA). The containment has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects the majority of the containment leakage that may occur following a loss of coolant accident (LOCA). The SBVS establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. Shield building Operability is required to ensure retention of primary containment leakage and proper operation of the SBVS. TS 3.6.10 is applicable in Modes 1, 2, 3, and 4.

The above TS are related to assuring the appropriate functional capability of plant equipment associated with containment systems required for safe operation of the facility and accident mitigation only when the reactor is in Modes 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address containment systems, are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.6 is acceptable.

T36.4 SUMMARY

Current Operating Licensed Condition

None of these TS currently apply with the reactor defueled.

Permanently Defueled Condition

Since these TS do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.6 is acceptable.

► TS SECTION 3.7, PLANT SYSTEMS ◄

T37.1 DESCRIPTION

The existing TS Section 3.7, "Plant Systems," contains Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of plant equipment required for safe operation of the facility, including the plant being in a defueled condition. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion.

T37.2 PROPOSED CHANGE

TS Section 3.7, Plant Systems

The following TS in Section 3.7 are being proposed for deletion, as identified in the table below: 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, 3.7.7, 3.7.8, 3.7.9, 3.7.10, 3.7.11, 3.7.12 and 3.7.16.

TS being retained and revised are 3.7.13, 3.7.14, and 3.7.15 as further described below and shown in Attachment 2.

TS Being Deleted		TS Being Revised
3.7	**PLANT SYSTEMS**	
3.7.1	Main Steam Safety Valves (MSSVs)	
3.7.2	Main Steam Isolation Valves (MSIVs)	
3.7.3	Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs), and MFRV Bypass Valves	
3.7.4	Steam Generator (SG) Power Operated Relief Valves (PORVs)	
3.7.5	Auxiliary Feedwater (AFW) System	
3.7.6	Condensate Storage Tanks (CSTs)	
3.7.7	Component Cooling (CC) System	
3.7.8	Service Water (SW) System	
3.7.9	Ultimate Heat Sink (UHS)	
3.7.10	Control Room Post Accident Recirculation (CRPAR) System	
3.7.11	Control Room Air Conditioning (CRAC) Alternate Cooling System	

TS Being Deleted	TS Being Revised
3.7.12 Auxiliary Building Special Ventilation (ASV) System	
	3.7.13 Spent Fuel Pool Water Level
	3.7.14 Spent Fuel Pool Boron Concentration
	3.7.15 Spent Fuel Pool Storage
3.7.16 Secondary Specific Activity	

The corresponding TS Bases sections are also being deleted or revised to reflect this change, as appropriate.

T37.3 TECHNICAL ANALYSIS

Section 3.7 TS That Are Not Applicable When Defueled

TS 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, 3.7.7, 3.7.8, 3.7.9, 3.7.12 and 3.7.16 do not currently apply with the reactor defueled.

TS 3.7.1, "Main Steam Safety Valves (MSSVs)," specifies requirements for the MSSVs to ensure they are capable of providing overpressure protection for the secondary system. TS 3.7.1 is applicable in Modes 1, 2, and 3.

TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," specifies requirements for the MSIVs to ensure that they are capable of isolating steam flow from the secondary side of the steam generators following a main steam line break (MSLB). TS 3.7.2 is applicable in Mode 1 and also applicable in Modes 2 and 3 except when all MSIVs are closed and deactivated.

TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)," Main Feedwater Regulation Valves (MFRVs), and MFRV Bypass Valves, specifies requirements for these respective components. The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety function of the MFRVs is to provide isolation of MFW flow to the secondary side of the steam generators following a main steam line break or an HELB. Closure of the MFIVs, or MFRVs and MFRV bypass valves, terminates flow to the steam generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs or MFRVs. TS 3.7.3 is applicable in Modes 1, 2, and 3 except when all MFIVs, MFRVs, and MFRV bypass valves are closed and deactivated or isolated by a closed manual valve.

TS 3.7.4, "Steam Generator (SG) Power Operated Relief Valves (PORVs)," specifies requirements for providing a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser not be available. TS 3.7.4 is applicable in Modes 1, 2, and 3, and also applicable in Mode 4 when steam generator is relied upon for heat removal.

TS 3.7.5, "Auxiliary Feedwater (AFW) System," specifies requirements to ensure that the AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. TS 3.7.5 is applicable in Modes 1, 2, and 3, and also applicable in Mode 4 when steam generator is relied upon for heat removal.

TS 3.7.6, "Condensate Storage Tanks (CSTs)," specifies requirements to ensure that two CSTs provide the preferred source of water to the steam generators for removing decay and sensible heat from the RCS. The CSTs provide a passive flow of water, by gravity, to the AFW System (TS 3.7.5). TS 3.7.6 is applicable in Modes 1, 2, and 3, and also applicable in Mode 4 when steam generator is relied upon for heat removal.

TS 3.7.7, "Component Cooling (CC) System," specifies requirements to ensure that the CC System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CC System also provides this function for various nonessential components. The CC System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment. TS 3.7.7 is applicable in Modes 1, 2, 3, and 4.

TS 3.7.8, "Service Water (SW) System," specifies requirements to ensure that the SW System provides a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. During normal operation, and a normal shutdown, the SW System also provides this function for various safety related and nonsafety related components. The safety related function is covered by this TS. TS 3.7.8 is applicable in Modes 1, 2, 3, and 4.

TS 3.7.9, "Ultimate Heat Sink (UHS)," specifies requirements to ensure that the UHS (Lake Michigan) provides a heat sink for process and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Circulating Water (CW) System, SW System, and the CC Water System. TS 3.7.9 is applicable in Modes 1, 2, 3, and 4.

TS 3.7.12, "Auxiliary Building Special Ventilation (ASV) System," specifies requirements to ensure that the ASV System filters air from the Containment and the Auxiliary Building (the ASV boundary). TS 3.7.12 is applicable in Modes 1, 2, 3, and 4.

TS 3.7.16, "Secondary Specific Activity," specifies a limit on secondary coolant specific activity during power operation, so as to minimize releases to the environment because of normal operation, anticipated operational occurrences, and accidents. TS 3.7.16 is applicable in Modes 1, 2, 3, and 4.

The above TS are related to assuring the appropriate functional capability of plant equipment required for safe operation of the facility only when the reactor is in Modes 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, are no longer applicable. Based on the above, the proposed deletion of TS related to these systems is acceptable.

TS 3.7.10, Control Room Post Accident Recirculation (CRPAR) System

TS 3.7.10, "Control Room Post Accident Recirculation (CRPAR) System," specifies requirements to ensure that the CRPAR System provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CRPAR System consists of two independent, redundant trains that recirculate and filter the control room and outside air. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Common ductwork, valves or dampers, and instrumentation also form part of the system.

The CRPAR System is an emergency system, which is normally in the standby mode of operation. The CRPAR System is part of the Control Room Air Conditioning (CRAC) System. During normal unit operation, the CRAC System provides cooling of recirculated and fresh air to ventilate the control room.

The neutral pressure envelope design of the control room minimizes infiltration of unfiltered air from the surrounding areas of the building. The CRPAR System fans are started upon receipt of a safety injection signal or high radiation signal as detected by the radiation monitor R-23 mounted in the main control room emergency zone (CREZ) supply duct.

The CRPAR System components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CRPAR System provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the USAR, Chapter 14.

TS 3.7.10 is applicable in Modes 1, 2, 3, 4, 5, and 6. It is also applicable during movement of irradiated fuel assemblies.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.7.10 is no longer needed for assuring the appropriate functional capability of the CRPAR System for safe operation of the facility when the reactor is in Modes 1 through 6. The only remaining TS 3.7.10 Applicability requirement for the CRPAR system is during movement of irradiated fuel assemblies.

The design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building, as discussed in Section 5.2 of this proposed amendment (Applicable Regulatory Requirements/Criteria). A description of the FHA analysis for the permanently defueled condition is provided in Section 2.0, "Proposed Change." The FHA analysis shows that the dose consequences are acceptable without relying on any SSCs to remain functional (including the CRPAR System) during and following the event (following 90 days of irradiated fuel decay time after reactor shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.13). As such, the CRPAR system is not required for providing airborne radiological protection for the control room operators. Consequently, the CRPAR System is not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA.

The requirement for the CRPAR system was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on the CRPAR system for accident mitigation (including any need for providing airborne radiological protection), the CRPAR 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 CRPAR system is being deleted because there are no design basis events that rely on the CRPAR system for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Although the CRPAR system also provides the primary means to ensure control room habitability in the event of a waste gas decay tank rupture accident in Modes 5 and 6, generation of radioactive waste gases has ceased since the permanent cessation of reactor operation. Since the waste gas decay tanks and volume control tanks will be purged of their contents, a rupture of these tanks is no longer an applicable accident.

Based on the above, the proposed deletion of TS 3.7.10 for the CRPAR System is acceptable.

TS 3.7.11, Control Room Air Conditioning (CRAC) Alternate Cooling System

TS 3.7.11, "Control Room Air Conditioning (CRAC) Alternate Cooling System," specifies requirements to ensure that the CRAC Alternate Cooling System provides temperature control for the control room following isolation of the control room during a design basis accident.

The CRAC Alternate Cooling System consists of two independent and redundant trains that provide cooling of recirculated and fresh air. Each train consists of an air handling unit (AHU) (containing filters, a cooling coil, and a fan), instrumentation, and controls to provide for control room temperature control.

The CRAC Alternate Cooling System is an emergency system, parts of which also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between 60°F and 85°F during normal operation using the non-safety related chiller. Under accident conditions (i.e., the non-safety related chillers not in service), cooling from the service water aligned directly to the AHU cooling coils will maintain temperature habitability of the control room environment and will maintain environment temperature for equipment operation. With a service water temperature of 80°F and a 95°F air ambient temperature, each CRAC Alternate Cooling train can maintain control room air temperature within the 110°F design temperature limit.

The design basis of the CRAC Alternate Cooling System is to maintain the control room temperature for 30 days of continuous operation.

During emergency operation, the CRAC Alternate Cooling System maintains the temperature < 110°F. A single active failure of a component of the CRAC Alternate Cooling System, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CRAC Alternate Cooling System is designed in accordance with Nuclear Safety Design Class I requirements. The CRAC Alternate Cooling System is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment Operability.

TS 3.7.11 is applicable in Modes 1, 2, 3, and 4. It is also applicable during movement of irradiated fuel assemblies.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.7.11 is no longer needed for assuring the appropriate functional capability of the CRAC Alternate Cooling System for safe operation of the facility when the reactor is in Modes 1 through 4.

The design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building, as discussed in Section 5.2 of this proposed amendment (Applicable Regulatory Requirements/Criteria). A description of the FHA analysis for the permanently defueled condition is provided in Section 2.0, "Proposed Change". The FHA analysis shows that following 90 days of decay time after reactor shutdown, the dose consequences of a postulated FHA (provided the spent fuel pool water level requirements of TS 3.7.13 are met) are acceptable without relying on any SSCs to remain functional during and following the event. Therefore, the CRAC Alternate Cooling System is not needed to support any SSCs relied upon for accident mitigation, nor is it needed for accident mitigation.

The requirement for the CRAC Alternate Cooling System was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the CRAC Alternate Cooling System is no longer relied upon for accident mitigation, it is also 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 CRAC Alternate Cooling System is being deleted because there are no design basis events that rely on the CRAC Alternate Cooling System for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Based on the above, the proposed deletion of TS 3.7.11 for the CRAC Alternate Cooling System is acceptable.

TS 3.7.13, Spent Fuel Pool Water Level

TS 3.7.13, "Spent Fuel Pool Water Level," specifies requirements to ensure that the minimum water level in the spent fuel pool (the north pool, south pool, and canal pool) meets the assumptions of iodine decontamination factors following a fuel handling accident (FHA). The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (July 2000). The resultant two hour Total Effective Dose Equivalent (TEDE) dose per person at the exclusion area boundary is < 25% of the 10 CFR 50.67 limits.

According to the analysis of the postulated fuel handling accident currently discussed in the Bases for TS 3.7.13, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Regulatory Guide 1.183 (July 2000) can be used directly.

In the FHA analysis, the is defined as the dropping of a spent fuel assembly onto the spent fuel pool floor or the racks that hold the spent fuel such that the cladding of all the fuel rods in one assembly ruptures. The gap activity in the damaged rods is instantaneously released into the spent fuel pool. The release occurs under 23 ft of water, which acts as a filter. The activity released from the spent fuel pool then mixes with the fuel building atmosphere before being exhausted to the environment. The fuel building exhaust rate is established to complete the release in 2 hours as required by RG 1.183. Therefore, the existing 23 ft water level requirement for the spent fuel pool remains appropriate based on the FHA analysis.

TS 3.7.13 is applicable during movement of irradiated fuel assemblies in the spent fuel pool.

TS 3.7.13 is being retained in the permanently defueled TS with essentially no change. 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 the TS SECTION 3.0 justification (Section T3.3 above). With the deletion of LCO 3.0.3, this note is rendered moot.

Retaining TS 3.7.13, with the proposed change, continues to ensure appropriate requirements for spent fuel pool water level.

TS 3.7.14, Spent Fuel Pool Boron Concentration

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel pool is based on the use of unborated water, which maintains each separate pool and the transfer canal in a subcritical condition during normal operation with the three separate pools fully loaded.

TS 3.7.14, "Spent Fuel Pool Boron Concentration," specifies requirements to ensure that the spent fuel pool boron concentration is $\geq$ 240 ppm. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

Accident analyses have determined that a minimum of 240 ppm of boron is sufficient to ensure criticality does not occur during the worst case fuel loading accident in the spent fuel pool racks (i.e., a fuel assembly misloaded in the canal pool racks). The analyses have also determined that no boron is necessary to ensure subcriticality during a dropped fuel assembly event in the spent fuel pool.

TS LCO 3.7.14 applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool.

TS 3.7.14 is being retained in the permanently defueled TS with essentially no change. The Note preceding 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 the TS SECTION 3.0 justification (Section T3.3 above). With the deletion of LCO 3.0.3, this note is rendered moot.

Retaining TS 3.7.14, with the proposed change, continues to ensure appropriate requirements for spent fuel pool boron concentration.

TS 3.7.15, Spent Fuel Pool Storage

The spent fuel pool at KPS is comprised of three separate pools, a large south pool, a smaller north pool, and a third pool designated as the canal pool and a fuel transfer canal that are connected to one another to allow for movement of spent fuel. The design of the spent fuel pool is based on the use of unborated water, which maintains each separate pool and the transfer canal in a subcritical condition during normal operation with the three separate pools fully loaded. The most severe accident scenario is the inadvertent placement of a fresh (unirradiated) fuel assembly into a location restricted to a burned assembly. This could potentially increase the reactivity of the north and south combined pools. To mitigate this postulated criticality related accident, boron is dissolved in the pool water. Safe operation of all three separate pools with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO.

TS 3.7.15, "Spent Fuel Pool Storage," specifies restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.15-1 in the accompanying LCO to ensure the k_{eff} of the spent fuel pool will always remain < 0.95, assuming the pool to be flooded with unborated water. Irradiated fuel assemblies discharged prior to or during the 1984 refueling outage with a combination of burnup and initial nominal enrichment in the Acceptable Domain of Figure 3.7.15-1 are allowed to be stored in the transfer canal spent fuel pool or the north and south combined spent fuel pools. New fuel assemblies, irradiated fuel assemblies discharged after the 1984 refueling outage, or spent fuel assemblies not in the Acceptable Domain of Figure 3.7.15-1 shall be stored in the north and south pools (combined).

TS LCO 3.7.15 applies whenever any fuel assembly is stored in the spent fuel pool.

TS 3.7.15 is being retained in the permanently defueled TS with only minor changes. 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 the TS SECTION 3.0 justification (Section T3.3 above). With the deletion of LCO 3.0.3, this note is rendered moot.

The phrase "New fuel assemblies" is being deleted from LCO 3.7.15.b. New fuel is no longer stored onsite and License Condition 2.B.(2) is being revised to no longer allow receipt of new fuel. Therefore, the phrase "New fuel assemblies" is no longer applicable and may be deleted.

Retaining TS 3.7.15, with the proposed change, continues to ensure appropriate requirements for controlling the movement of each fuel assembly in the spent fuel pool and managing the storage location of each fuel assembly.

T37.4 SUMMARY

Current Operating Licensed Condition

TS 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, 3.7.7, 3.7.8, 3.7.9, 3.7.12, and 3.7.16 do not currently apply with the reactor defueled.

TS 3.7.10, 3.7.11, 3.7.13, 3.7.14, and 3.7.15 remain applicable with the reactor defueled.

Permanently Defueled Condition

Since TS 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, 3.7.7, 3.7.8, 3.7.9, 3.7.12, and 3.7.16 do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.7.10 and 3.7.11 are not needed for accident mitigation in the permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.7.13, 3.7.14, and 3.7.15 will remain applicable with the reactor permanently defueled. As such, they are being retained and revised to reflect a permanently defueled condition.

Conclusion

Deleting TS 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, 3.7.7, 3.7.8, 3.7.9, 3.7.10, 3.7.11, 3.7.12, and 3.7.16 in Section 3.7 is acceptable.

Retaining TS 3.7.13, 3.7.14, and 3.7.15, as revised, ensures appropriate functional capability requirements for the associated equipment.

► TS SECTION 3.8, ELECTRICAL POWER SYSTEMS ◄

T38.1 DESCRIPTION

The existing TS Section 3.8, "Electrical Power Systems," contains Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility, including the plant being in a defueled condition. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply (or are no longer needed) in a defueled condition are being proposed for deletion.

T38.2 PROPOSED CHANGE

TS Section 3.8, Electrical Power Systems

All TS in Section 3.8 are being proposed for deletion, as identified in the table below.

TS Being Deleted	TS Being Revised
3.8 ELECTRICAL POWER SYSTEMS	
3.8.1 AC Sources - Operating	
3.8.2 AC Sources - Shutdown	
3.8.3 Diesel Fuel Oil and Lube Oil	
3.8.4 DC Sources - Operating	
3.8.5 DC Sources - Shutdown	
3.8.6 Battery Parameters	
3.8.7 Inverters - Operating	
3.8.8 Inverters - Shutdown	
3.8.9 Distribution Systems - Operating	
3.8.10 Distribution Systems - Shutdown	

The corresponding TS Bases sections are also being deleted to reflect this change.

T38.3 TECHNICAL ANALYSIS

Section 3.8 TS That Are Not Applicable When Defueled

TS 3.8.1, 3.8.4, 3.8.7, and 3.8.9 do not currently apply with the reactor defueled.

TS 3.8.1, "AC Sources – Operating," specifies requirements to ensure that the offsite power sources (reserve auxiliary transformer and tertiary auxiliary transformer), and the onsite standby power sources (emergency diesel generators (EDGs) A and B), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature (ESF) systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. TS 3.8.1 is applicable in Modes 1, 2, 3, and 4.

TS 3.8.4, "DC Sources – Operating," specifies requirements to ensure that the DC electrical power subsystems (with each subsystem consisting of one battery, either battery charger for each battery, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem) are required to be Operable to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident (DBA).

TS 3.8.7, "Inverters – Operating," specifies requirements to ensure that required inverters are Operable such that the redundancy incorporated into the design of the reactor protection system (RPS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation and controls is maintained. These requirements include the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, RCS, and containment design limits are not exceeded.

TS 3.8.9, "Distribution Systems – Operating," specifies requirements to ensure availability of AC, DC, and AC instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The AC, DC, and AC instrument bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

The preceding TS are related to assuring the appropriate functional capability of plant equipment required for safe operation of the facility only when the reactor is in Modes 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, are no longer applicable. Based on the above, the proposed deletion of TS related to these systems is acceptable.

TS 3.8.2, AC Sources - Shutdown

TS 3.8.2, "AC Sources – Shutdown," specifies requirements to ensure that the offsite power sources (reserve auxiliary transformer and tertiary auxiliary transformer), and the onsite standby power sources (emergency diesel generators (EDGs) A and B), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature (ESF) systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded.

The Operability of the minimum AC sources during Modes 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) have no specific analyses in Modes 5 and 6. Worst case bounding events are deemed not credible in Modes 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

TS 3.8.2 is applicable in Modes 5 and 6. It is also applicable during movement of irradiated fuel assemblies.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.2 is no longer needed for assuring the appropriate functional capability of the AC sources for safe operation of the facility when the reactor is in Modes 5 and 6. The only remaining TS 3.8.2 Applicability requirement for AC sources is during movement of irradiated fuel assemblies.

The design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building (as discussed in Section 5.2, "Applicable

Regulatory Requirements/Criteria," of this proposed amendment). A description of the FHA analysis for the permanently defueled condition is provided in Section 2.0, "Proposed Change." The FHA analysis shows that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event (following 90 days of irradiated fuel decay time after reactor shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.13).

The requirement for AC sources was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The FHA is the applicable design basis accident related to the TS requirement for functional capability of AC sources (offsite power and EDG) during the TS specified condition of "During movement of irradiated fuel assemblies". Because the FHA analysis does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of irradiated fuel assemblies for mitigation of a potential FHA. Therefore, during movement of irradiated fuel assemblies, 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 AC sources is being deleted because there are no design basis events that rely on AC sources for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

With the reactor permanently defueled, irradiated fuel is stored either in the independent spent fuel storage facility (ISFSI) or in the spent fuel pool. The ISFSI is a passive system that does not rely on electrical power for heat transfer. Since there is a large capacity for heat absorption in the spent fuel pool, active system components are not redundant. Alternate cooling capability can be made available under anticipated malfunctions or failures without reliance on EDGs.

The existing requirement for a qualified offsite circuit is based on its need to be capable of maintaining rated frequency and voltage, accepting required loads during an accident, and capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown". Because the requirement for ESF equipment no longer exists upon permanent defueling (as justified in the associated sections of this proposed amendment), there is no longer a need for ESF buses. Since the AC electrical power distribution subsystems is no longer required to power ESF equipment, TS 3.8.10 is being deleted, as described in the corresponding section below.

With no need for a qualified offsite circuit to be capable of supplying loads during an accident, while connected to the ESF buses; and no need for an EDG, there is no longer a need for TS 3.8.2. Therefore, TS 3.8.2 is being deleted in its entirety.

Because of the continued need for normal electrical power to supply equipment needed for cooling of irradiated fuel stored in the spent fuel pool, and as defense in depth for ensuring normal electrical power availability during a response to a FHA, the pertinent requirements of TS 3.8.2 are being relocated to the Technical Requirements Manual (TRM). The TRM (which is part of the USAR and therefore subject to the requirements of 10 CFR 50.59) will retain a requirement for only a single Functional circuit between the offsite transmission network and the (non-Class 1E) onsite AC electrical power distribution subsystems. There is no need for emergency power in order to fulfill the SFP cooling function. The requirement for a single circuit between the offsite transmission network and the onsite AC distribution subsystems will be applicable whenever any irradiated fuel assembly is stored in the spent fuel pool and also applicable during movement of irradiated fuel assemblies.

Existing Required Action A.2.2 (Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration) is not being relocated to the TRM. Loss of required SDM (Mode 5) or boron concentration (Mode 6) pertains to operation of the RCS, which is no longer applicable in a permanently defueled condition.

Existing Condition B and its associated Required Action and Completion Time are not being relocated to the TRM so as to conform to the elimination of the EDG requirement.

The existing SR to verify correct breaker alignment and indicated power availability for the offsite circuit, at a seven day Frequency, remains applicable and is being relocated to the TRM.

Relocating the pertinent functionality requirements of TS 3.8.2 into the TRM continues to ensure appropriate requirements for AC sources. Additionally, the TRM currently contains requirements for the Technical Support Center (TSC) diesel generator (DG), which apply at all times, and which will remain in effect subject to the requirements of 10 CFR 50.59. The TSC DG is an independent, non-class 1E, 600 kW power source that provides AC power to onsite 480 V buses. The TSC DG starts automatically on loss of voltage to its associated 480 V bus so as to supply power to the bus. The TSC DG provides power to the TSC Building, security lighting system and other non-ESF plant systems that are required to operate upon loss of off-site electrical sources.

This change is consistent with the associated change approved by NRC for Millstone Power Station Unit 1 in License Amendment 106, dated November 9, 1999 (Reference 6); and for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively), dated December 30, 1999 (Reference 7). The license amendments for both Millstone and Zion relaxed all TS requirements for AC sources.

TS 3.8.3, Diesel Fuel Oil and Lube Oil

For proper operation of the standby DGs, it is necessary to ensure sufficient quantity and proper quality of the fuel oil as well as sufficient quantity of lube oil. TS 3.8.3, "Diesel Fuel Oil and Lube Oil," specifies these parameters for stored diesel fuel oil and lube oil. Stored diesel fuel oil is required to have sufficient supply for 7 days of rated load operation for each DG. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate each DG at rated load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated design basis accident (DBA) with loss of offsite power.

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil and lube oil support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil and lube oil are required to be within limits when the associated DG is required to be Operable. As such TS LCO 3.8.3 is applicable when the associated DG is required to be Operable.

As discussed in the justification for deleting TS 3.8.2 above, the requirement for EDGs is being deleted from the TS because there are no design basis accidents or transients analyzed in USAR Chapter 14 that rely on the EDGs for mitigation. Since TS 3.8.3 exists solely to support the EDG requirements of TS 3.8.1 and 3.8.2, the elimination of the need for EDGs also obviates the need for their support systems. As such, TS 3.8.3 may be deleted.

Based on the above, the proposed deletion of TS 3.8.3 for fuel oil and lube oil parameters is acceptable.

TS 3.8.5, DC Sources - Shutdown

The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all Modes of operation.

TS 3.8.5, "DC Sources – Shutdown," specifies requirements to ensure that the DC electrical power subsystems (with each subsystem consisting of one battery, either battery charger for each battery, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem) are required to be Operable to support one subsystem of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (i.e., fuel handling accidents and inadvertent dilution events).

TS 3.8.5 is applicable in Modes 5 and 6. It is also applicable during movement of irradiated fuel assemblies. One DC electrical power source is required to be Operable in Modes 5 and 6, and during movement of irradiated fuel assemblies, to provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.5 is no longer needed for assuring the appropriate functional capability of the DC sources for safe operation of the facility when the reactor is in Modes 5 or 6.

The design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building, as discussed in Section 5.2 of this proposed amendment (Applicable Regulatory Requirements/Criteria). A description of the FHA analysis for the permanently defueled condition is provided in Section 2.0, "Proposed Change." Because the FHA analysis does not rely on DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), DC sources are therefore not required for accident mitigation. Consequently, DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA. Thus, the requirement for DC sources is being deleted.

The requirement for DC sources was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on DC sources for accident mitigation, DC sources are 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 DC sources is being deleted because there are no design basis accidents or transients analyzed in USAR Chapter 14 that rely on DC sources for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Based on the above, the proposed deletion of TS 3.8.5 for DC sources is acceptable.

TS 3.8.6, Battery Parameters

TS 3.8.6, "Battery Parameters," delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC power subsystem batteries. In addition to the limitations of this Specification, the Battery Monitoring and Maintenance Program also implements a program specified in Specification 5.5.15 for monitoring various battery parameters. Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident.

Battery parameters are required solely for the support of the associated DC electrical power subsystems (per TS LCO 3.8.4 and LCO 3.8.5). Therefore, battery parameter limits are only required (and TS 3.8.6 is only applicable) when the DC electrical power source is required to be Operable.

As discussed in the justification for deleting TS 3.8.5 above, the requirement for DC sources is being deleted from the TS because there are no design basis accidents and transients analyzed in USAR Chapter 14 that rely on the DC sources for mitigation. Since TS 3.8.6 exists solely to support the DC source requirements of TS 3.8.4 and 3.8.5, the elimination of the need for DC sources also obviates the need for their support systems. As such, TS 3.8.6 may be deleted.

Based on the above, the proposed deletion of TS 3.8.6 for battery parameters is acceptable.

TS 3.8.8, Inverters - Shutdown

The inverters are the preferred source of power for the AC instrument buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the instrument buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The inverter provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System

(RPS) and the Engineered Safety Feature Actuation System (ESFAS) so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded.

The Operability of one inverter to a required 120 VAC instrument bus during Modes 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many design basis accidents have no specific analyses in Modes 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

TS 3.8.8, "Inverters – Shutdown," specifies requirements to ensure the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (i.e., fuel handling accidents and inadvertent dilution events).

TS 3.8.8 is applicable in Modes 5 and 6. It is also applicable during movement of irradiated fuel assemblies. The inverter that is required to be Operable in Modes 5 and 6 and during movement of irradiated fuel assemblies provides assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
b. Systems needed to mitigate a fuel handling accident are available;
c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and,
d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.8 is no longer needed for assuring the appropriate functional capability of inverters for safe operation of the facility when the reactor is in Modes 5 or 6.

The design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building, as discussed in Section 5.2 of this proposed amendment (Applicable Regulatory Requirements/Criteria). A description of the FHA analysis for the permanently defueled condition is provided in Section 2.0, "Proposed Change." Because the FHA analysis does not rely on inverters for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), inverters are therefore not required for accident mitigation. Consequently, inverters are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA.

The requirement for inverters was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on inverters for accident mitigation, the inverters are 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 inverters is being deleted because there are no design basis accidents or transients analyzed in USAR Chapter 14 that rely on the inverters for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Based on the above, the proposed deletion of TS 3.8.8 for inverters is acceptable.

TS 3.8.10, Distribution Systems - Shutdown

TS 3.8.10, "Distribution Systems – Shutdown," specifies requirements to ensure that the onsite AC, DC, and AC instrument bus electrical power distribution systems provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the minimum AC, DC, and AC instrument bus electrical power distribution subsystems during Modes 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and,

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

This TS explicitly requires energization of the portions of the electrical power distribution system necessary to support OPERABILITY of required systems, equipment, and components. Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

TS 3.8.10 is applicable in Modes 5 and 6. It is also applicable during movement of irradiated fuel assemblies.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.10 is no longer needed for assuring the appropriate functional capability of the electrical distribution systems for safe operation of the facility when the reactor is in Modes 5 and 6.

The design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the fuel handling accident (FHA) in the auxiliary building (as discussed in Section 5.2, Applicable Regulatory Requirements/Criteria, of this proposed amendment). A description of the FHA analysis for the permanently defueled condition is provided in Section 2.0, "Proposed Change." The FHA analysis shows that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event (following 90 days of irradiated fuel decay time after reactor shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.13).

The requirement for electrical distribution systems was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on electrical distribution systems for accident mitigation, electrical distribution systems are 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 electrical distribution systems is being deleted because there are no design basis events that rely on electrical distribution systems for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

The existing requirement for onsite Class 1E electrical power distribution subsystems of LCO 3.8.10 is to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. Because there is no longer a need for any ESF systems for accident mitigation, the requirements of TS 3.8.10 are no longer needed. Therefore, TS 3.8.10 is being deleted in its entirety.

Because of the continued need for electrical power to supply equipment needed for cooling of irradiated fuel stored in the spent fuel pool and as defense in depth for mitigation of a FHA, the pertinent requirements of TS 3.8.10 are being relocated to the Technical Requirements Manual (TRM). The TRM will retain a requirement that the necessary portion of electrical power distribution subsystems shall be Functional to support equipment required to be Functional. This requirement will be applicable whenever any irradiated fuel assembly is stored in the spent fuel pool and also during movement of irradiated fuel assemblies.

Existing Required Action A.2.2 (Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration) is not being relocated to the TRM. Loss of required SDM (Mode 5) or boron concentration (Mode 6) pertains to operation of the RCS, which is no longer applicable in a permanently defueled condition.

Existing Required Action A.2.4 (Declare associated required residual heat removal subsystem(s) inoperable and not in operation) is not being relocated to the TRM. The requirement for residual heat removal subsystems is no longer applicable in a permanently defueled condition. Deletion of the requirement for residual heat removal (TS Section 3.5) is described in the TS SECTION 3.5 justification (Section T35.3 above). With the deletion of TS Section 3.5, Required Action A.2.4 of TS 3.8.10 is rendered moot.

The existing SR to verify correct breaker alignments and voltage to required electrical power distribution subsystems, at a seven day Frequency, remains applicable and is being relocated to the TRM.

Relocating the pertinent functionality requirements of TS 3.8.10 into the TRM continues to ensure appropriate requirements for onsite electrical distribution systems.

This change is consistent with the associated change approved by NRC for Millstone Power Station Unit 1 in License Amendment 106, dated November 9, 1999 (Reference 6); and for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively), dated December 30, 1999 (Reference 7). The license amendments for both Millstone and Zion completely relaxed all TS requirements for onsite electrical distribution systems.

T38.4 SUMMARY

Current Operating Licensed Condition

TS 3.8.1, 3.8.4, 3.8.7, and 3.8.9 do not currently apply with the reactor defueled.

TS 3.8.2, 3.8.3, 3.8.5, 3.8.6, 3.8.8, and 3.8.10 remain applicable with the reactor defueled.

Permanently Defueled Condition

Since TS 3.8.1, 3.8.4, 3.8.7, and 3.8.9 do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.8.2, 3.8.3, 3.8.5, 3.8.6, 3.8.8 and 3.8.10 are not needed for accident mitigation in the permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

The pertinent requirements of TS 3.8.2 and 3.8.10 that are relied on for functionality of irradiated fuel cooling and as defense in depth for mitigation of a fuel handling accident are being relocated to the Technical Requirements Manual.

Conclusion

Deleting all TS in Section 3.8, while retaining pertinent requirements of TS 3.8.2 and 3.8.10 in the Technical Requirements Manual is acceptable.

► TS SECTION 3.9, REFUELING OPERATIONS ◄

T39.1 DESCRIPTION

The existing TS Section 3.9, "Refueling Operations," contains Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of parameters and equipment within containment that are required for mitigation of design basis accidents during refueling operations (moving irradiated fuel to or from the reactor core). Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion.

T39.2 PROPOSED CHANGE

TS Section 3.9, Refueling Operations

All TS in Section 3.9 are being proposed for deletion, as identified in the table below.

TS Being Deleted		TS Being Revised
3.9	REFUELING OPERATIONS	
3.9.1	Boron Concentration	
3.9.2	Nuclear Instrumentation	
3.9.3	Residual Heat Removal (RHR) and Coolant Circulation – High Water Level	
3.9.4	Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level	
3.9.5	Refueling Cavity Water Level	
3.9.6	Containment Penetrations	

The corresponding TS Bases sections are also being deleted to reflect this change.

T39.3 TECHNICAL ANALYSIS

Section 3.9 TS That Are Not Applicable When Defueled

None of the TS in Section 3.9 currently apply with the reactor permanently defueled and fuel being prohibited from being placed in the reactor vessel per 10 CFR 50.82(a)(2).

TS 3.9.1, "Boron Concentration," places limits on the boron concentrations of the Reactor Coolant System (RCS), the fuel transfer canal, the refueling cavity, and the spent fuel pool during refueling, to ensure that the reactor remains subcritical during Mode 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. The boron concentration limits required by TS LCO 3.9.1 are specified in the Core Operating Limits Report (COLR). The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. TS 3.9.1 is applicable in Mode 6.

TS 3.9.2, "Nuclear Instrumentation," requires that two source range neutron flux monitors be Operable to ensure that redundant monitoring capability is available to detect changes in core reactivity. In addition, at least one of the two monitors must provide an Operable audible count rate function in the control room to alert the operators to the initiation of a boron dilution event. Two Operable source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident or an improperly loaded fuel assembly. The audible count rate from the source range neutron flux monitors provides prompt and definite indication of any boron dilution. TS 3.9.2 is applicable in Mode 6.

TS 3.9.3, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," specifies requirements for the RHR System in Mode 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, and to prevent boron stratification. One loop of the RHR System is required to be Operable and in operation in Mode 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be Operable, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. TS 3.9.3 is applicable in Mode 6, with the water level ≥ 23 ft above the top of the reactor vessel flange.

TS 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level," also specifies requirements for the RHR System in Mode 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, and to prevent boron stratification. However, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be Operable. Additionally, one loop of RHR must be in operation. TS 3.9.4 is applicable in Mode 6, with the water level < 23 ft above the top of the reactor vessel flange.

TS 3.9.5, "Refueling Cavity Water Level," specifies a minimum water level of 23 ft above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits. The requirements of TS LCO 3.9.5, in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in Regulatory Guide 1.183. TS 3.9.5 is only applicable during movement of irradiated fuel assemblies within containment.

TS 3.9.6, "Containment Penetrations," specifies requirements for containment closure during the conduct of refueling operations. The containment penetrations included within this TS are the equipment hatch, personnel air lock doors, and penetrations that provide direct access from the containment atmosphere to the outside atmosphere. This LCO limits the consequences of a fuel handling accident involving handling irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. TS 3.9.6 is only applicable during movement of irradiated fuel assemblies within containment.

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe refueling operation of the facility only when the reactor is in Mode 6 or during movement of irradiated fuel assemblies within containment. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, which thereby precludes entry into Mode 6. The prohibition on placing fuel in the reactor vessel also precludes movement of fuel within containment. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, control of process variables, design features, or operating restrictions are no longer applicable.

Based on the above, the proposed deletion of all TS in Section 3.9 is acceptable.

T39.4 SUMMARY

Current Operating Licensed Condition

TS 3.9.1, 3.9.2, 3.9.3 and 3.9.4 do not currently apply with the reactor defueled.

TS 3.9.5 and 3.9.6 remain applicable with the reactor defueled, but only during movement of irradiated fuel assemblies within containment.

Permanently Defueled Condition

Since TS 3.9.1, 3.9.2, 3.9.3 and 3.9.4 do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Since TS 3.9.5 and 3.9.6 do not apply with no movement of irradiated fuel assemblies within containment, they also are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting all TS in Section 3.9 is acceptable.

► TS SECTION 4.0, DESIGN FEATURES ◄

T4.1 DESCRIPTION

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, would have a significant effect on safety and are not covered in the previous sections of the TS. Because the KPS Part 50 license no longer authorizes 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.

T4.2 PROPOSED CHANGE

TS Section 4.0, Design Features

The following TS in Section 4.0 is being proposed for deletion, as identified in the table below: 4.2. TS being retained and revised are 4.1 and 4.3, as further described below and shown in Attachment 2.

TS Being Deleted	TS Being Revised
4.0 DESIGN FEATURES	
	4.1 Site Location
4.2 Reactor Core	
	4.3 Fuel Storage

There are no corresponding TS Bases sections associated with this TS section.

T4.3 TECHNICAL ANALYSIS

TS 4.1, Site Location

TS 4.1, "Site Location," provides a description regarding the location of KPS. This TS section is being retained in the permanently defueled TS with no changes.

TS 4.2, Reactor Core

TS 4.2, "Reactor Core," provides a description and requirements regarding the reactor core fuel assemblies and control rod assemblies. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, this TS section does not apply in a defueled condition and is being proposed for deletion.

This TS section will read as follows.

4.2 Deleted.

TS 4.3, Fuel Storage

TS 4.3, "Fuel Storage," provides a description and requirements regarding prevention of criticality of spent fuel, prevention of spent fuel pool drainage, and spent fuel capacity limitations. This TS section is being retained as-is in the permanently defueled TS, with the exception of TS 4.3.1.2.

TS 4.3.1.2, regarding designed and maintenance of the new fuel storage racks, is being deleted. New fuel is no longer stored onsite and License Condition 2.B.(2) is being revised to no longer allow receipt of new fuel. Therefore, the design feature associated with the new fuel storage racks is no longer applicable and may be deleted.

This TS subsection will read as follows.

4.3.1.2 Deleted.

T4.4 SUMMARY

Current Operating Licensed Condition

TS Section 4.0, "Design Features," does not contain applicability requirements. As such, all parts of this section can be conservatively defined as being applicable at all times.

Permanently Defueled Condition

TS 4.1 will remain germane with the reactor permanently defueled. As such, this TS section is being retained to reflect a permanently defueled condition.

Since TS 4.2 contains requirements only associated with the reactor core, which can no longer be used following submittal of the certifications required by 10 CFR 50.82(a), this

TS section is not needed for a permanently defueled condition. 10 CFR 50.82(a)(2) prohibits DEK from operating the plant or placing fuel in the reactor vessel. Therefore, this TS section is no longer applicable. As such, TS 4.2 may be deleted with no impact on continued safe operation of the facility.

TS 4.3 (with the exception of TS 4.3.1.2) will remain germane with the reactor permanently defueled. As such, this TS section (with the exception of TS 4.3.1.2) is being retained to reflect a permanently defueled condition. Because License Condition 2.B.(2) is being revised to no longer allow receipt of new fuel, TS 4.3.1.2 may be deleted with no impact on continued safe operation of the facility.

Conclusion

Deleting TS 4.2 in Section 4.0 is acceptable.

Retaining TS 4.1 and TS 4.3, as revised, ensures appropriate requirements for the associated design features.

► TS SECTION 5.0, ADMINISTRATIVE CONTROLS ◄

T5.1 DESCRIPTION

The existing TS Section 5.0, "Administrative Controls," contains provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the administrative controls that do not apply in a defueled condition are being proposed for deletion.

T5.2 PROPOSED CHANGE

TS Section 5.0, Administrative Controls

All TS in Section 5.0 are being retained and revised, as identified in the table below and as further described and as shown in Attachment 2.

TS Being Deleted	TS Being Revised	
5.0 ADMINISTRATIVE CONTROLS		
	5.1	Responsibility
	5.2	Organization
	5.3	Unit Staff Qualifications
	5.4	Procedures
	5.5	Programs and Manuals
	5.6	Reporting Requirements
	5.7	High Radiation Area

There are no corresponding TS Bases sections associated with this TS section.

T5.3 TECHNICAL ANALYSIS

TS 5.1, Responsibility

TS 5.1, "Responsibility," provides a description and requirements regarding certain key operational management responsibilities. TS 5.1 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.1 includes certain requirements associated with reactor operation, which is no longer permitted following submittal of the certifications required by 10 CFR 50.82(a). Therefore, those requirements are not needed for a permanently defueled condition because 10 CFR 50.82(a)(2) prohibits DEK from operating the plant or placing fuel in the reactor vessel.

TS 5.1.1 is being revised to more appropriately refer to "plant" operation, rather than "unit" operation (the term "unit" is typically associated with the reactor).

Likewise, TS 5.1.2 is being revised to replace the term "control room" with "shift" to more appropriately reflect the defueled command function. Safe operation in the permanently defueled condition consists primarily of ensuring safe management of the irradiated fuel that is stored onsite. Associated activities (e.g., fuel handling) do not necessarily rely on the control room. With KPS permanently shutdown and defueled, the number of relevant controls located in the control room and the gradual nature of abnormal or accident situations would not warrant that the command function remain in the control room. Adequate communications capability is provided to allow operators to appropriately safely manage storage and handling of irradiated fuel without reliance on the control room for the command function.

The requirements of TS 5.1.2 associated with the unit being in Modes 1 through 6 are being deleted because they are not applicable with the reactor in the permanently defueled condition.

TS 5.2, Organization

TS 5.2, "Organization," provides a description and requirements regarding the facility organization. TS 5.2 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.2 includes certain requirements associated with reactor operation, which is no longer permitted following submittal of the certifications required by 10 CFR 50.82(a). Therefore, those requirements are not needed for a permanently defueled condition because 10 CFR 50.82(a)(2) prohibits DEK from operating the plant or placing fuel in the reactor vessel.

TS 5.2.1, "Onsite and Offsite Organization," is being revised to replace reference to terms such as "unit", "operation", and "power plant" with terms more appropriate to the defueled condition such as "plant", "storage", "management" and "nuclear fuel".

In TS 5.2.1.d, the term "the operating staff" is being replaced with "CERTIFIED FUEL HANDLERS", as defined in TS 1.1, Definitions. In the defueled condition, the primary responsibility of managing safe storage of the irradiated fuel will be performed by

Certified Fuel Handlers. Additionally, since the plant will not be permitted to operate, the phrase "independence from operating pressures" is being replaced with the phrase "ability to perform their assigned functions", which is an analogous requirement for the defueled condition.

TS 5.2.2, "Unit Staff," is being revised to replace reference to the term "unit" with "facility", because the term "unit" is typically associated with the reactor.

TS 5.2.2.a is being deleted because this requirement only applies in Modes 1, 2, 3, and 4 or if the reactor contains fuel. Therefore, it is not applicable with the reactor in the permanently defueled condition.

The requirement being deleted from TS 5.2.2.a is being replaced by a new requirement for each on duty shift to be composed of at least the minimum shift crew composition shown in new TS Table 5.2.2-1. This new Table specifies a minimum shift crew composition staffing requirement of one Certified Fuel Handler and one Non-Certified Operator. A footnote is added stating that a second Certified Fuel Handler may be used in lieu of a Non-Certified Operator. This minimum shift crew composition is appropriate for the safe management of irradiated fuel at a permanently defueled facility.

TS 5.2.2.b is being revised to replace references to 10 CFR 50.54(m)(2)(i) and TS 5.2.2.a and TS 5.2.2.e with reference to the new TS Table 5.2.2-1. KPS is not required to have operators licensed pursuant to part 55 following submittal of the certifications required by 10 CFR 50.82(a). As a result, 10 CFR 50.54(m)(2)(i) and TS 5.2.2.a and TS 5.2.2.e will not apply. Therefore, references to these superseded requirements are being changed to refer to the new minimum shift crew composition of TS Table 5.2.2-1.

TS 5.2.2.c is being deleted because this requirement only applies when fuel is in the reactor. Therefore, it is not applicable with the reactor in the permanently defueled condition.

The requirement being deleted from TS 5.2.2.c is being replaced by a new requirement specifying that all fuel handling operations shall be directly supervised by a qualified individual. This new requirement ensures that movement of irradiated fuel is only performed under the direct supervision of an individual who has been trained and qualified on the procedures, processes, requirements and standards for safe movement of irradiated fuel.

TS 5.2.2.d is being revised to replace the requirement that the operations manager or assistant operations manager shall hold a Senior Operator license, with a requirement that the shift manager shall be a Certified Fuel Handler. Since the certifications required by 10 CFR 50.82(a) have been submitted, the requirements of 10 CFR 50.54(m) are no longer applicable because the KPS Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel. These certifications also obviate the need for the operators' licenses specified in 10 CFR 55. Therefore, there is no longer a need for operations management staff to hold a Senior Operator license. Replacing this with a requirement that the shift manager shall be a Certified Fuel Handler ensures that the senior individual on shift is appropriately trained and qualified, in accordance with the NRC-approved Certified Fuel Handler training program, to supervise shift activities.

TS 5.2.2.e is being deleted because this requirement only applies in Modes 1, 2, 3, and 4. Therefore, it is not applicable with the reactor in the permanently defueled condition.

TS 5.3, Unit Staff Qualifications

TS 5.3, "Unit Staff Qualifications," provides a description and requirements regarding qualifications of the facility staff. TS 5.3 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.3 is being revised to replace reference to the term "unit" with "facility", because the term "unit" is typically associated with the reactor.

TS 5.3.1.b is being deleted because the requirements for licensed operators no longer apply following submittal of the certifications required by 10 CFR 50.82(a).

TS 5.3.2 is being deleted because neither 10 CFR 50.54(m) nor the requirement for licensed operators per 10 CFR 54 apply following submittal of the certifications required by 10 CFR 50.82(a).

A new TS 5.3.2 is being added to require that an NRC approved training and retraining program for the Certified Fuel Handlers (CFH) shall be maintained. The CFH Training Program ensures that the qualifications of fuel handlers are commensurate with the tasks to be performed and the conditions requiring response. 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," requires training programs to be derived using a systems approach to training (SAT) as defined in 10 CFR 55.4. Although the requirements of 10 CFR 50.120 apply to holders of an operating license issued under Part 50, and the DEK license no longer authorizes operation following submittal of the certifications required by 10 CFR 50.82(a), the CFH training program nonetheless aligns with those requirements. The CFH Training Program provides adequate confidence that appropriate SAT based training of personnel who will perform CFH duties is conducted to ensure the facility is maintained in a safe and stable condition.

TS 5.4, Procedures

TS 5.4, "Procedures," provides a description and requirements regarding administration of written procedures. TS 5.4 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.4.1.b is being deleted because the emergency operating procedures discussed therein are no longer required. These emergency operating procedures pertained only to events resulting from reactor operation. Therefore, they are not needed with the reactor in the permanently defueled condition.

TS 5.5, Programs and Manuals

TS 5.5, "Program and Manuals," provides a description and requirements regarding programs and manuals that are to be established, implemented, and maintained. TS 5.5 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.5.2, "Primary Coolant Sources Outside Containment," is being deleted because the System Integrity Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.4, "Component Cyclic or Transient Limit Program," is being deleted because the Component Cyclic or Transient Limit Program pertains only to reactor support systems that do not apply in a defueled condition.

TS 5.5.5, "Reactor Coolant Pump Flywheel Inspection Program," is being deleted because the Reactor Coolant Pump Flywheel Inspection Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.6, "Inservice Testing Program," is being deleted because the Inservice Testing Program is no longer required in a permanently defueled condition. There are no longer any ASME Code Class 1, 2, or 3 components that are required to perform a specific function in mitigating the consequences of an accident when in a permanently defueled condition.

TS 5.5.7, "Steam Generator (SG) Program," is being deleted because the Steam Generator Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.8, "Secondary Water Chemistry Program," is being deleted because the Secondary Water Chemistry Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.9, "Ventilation Filter Testing Program (VFTP)," is being deleted because the Ventilation Filter Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The accident analysis applicable to the permanently defueled condition does not rely on ventilation filters for accident mitigation.

TS 5.5.10, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," is being deleted because the Explosive Gas and Storage Tank Radioactivity Monitoring Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The tanks associated with this program will be vented and removed from service and there will no longer be any source of explosive or radioactive gases generated from reactor operation.

TS 5.5.11, "Diesel Fuel Oil Testing Program," is being deleted because the Diesel Fuel Oil Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirement for emergency diesel generators, which are supported by the fuel oil being tested per this program, is being deleted as described in preceding sections. The accident analysis applicable to the permanently defueled condition does not rely on emergency diesel generators for accident mitigation.

TS 5.5.13, "Safety Function Determination Program (SFDP)," is being deleted because the Safety Function Determination Program (SFDP) is not needed in a permanently defueled condition. 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. As discussed in Section 5.2 of this proposed amendment (Applicable Regulatory Requirements/Criteria), most of the design basis accidents and transients analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition. After the termination of reactor operations at KPS and the permanent removal of the fuel from the reactor vessel (following 90 days of decay time after shutdown), none of the systems, structures, and components (SSCs) at KPS are required to be relied on for accident mitigation. Therefore, none of the SSCs at KPS meet the definition of a safety-related SSC stated in 10 CFR 50.2 (with the exception of the passive spent fuel pool structure).

TS 5.5.14, "Containment Leakage Rate Testing Program," is being deleted because the Containment Leakage Rate Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirement for containment systems is being deleted as described in preceding sections. Therefore, the need for leakage rate testing of containment is no longer germane.

TS 5.5.15, "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 preceding sections. The accident analysis applicable to the permanently defueled condition does not rely on batteries for accident mitigation.

TS 5.5.16, "Setpoint Control Program," is being deleted because the Setpoint Control Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The five TS LCOs to which this program applies (listed in TS 5.5.16) are being deleted as described in preceding sections. Therefore the need for this program no longer exists.

TS 5.6, Reporting Requirements

TS 5.6, "Reporting Requirements," provides a description and requirements regarding reports that are to be submitted in accordance with 10 CFR 50.4. TS 5.6 will remain germane with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.6.1, "Annual Radiological Environmental Operating Report," and TS 5.6.2, "Radioactive Effluent Release Report," are being revised to replace reference to the term "unit" with "facility", because the term "unit" is typically associated with the reactor.

TS 5.6.3, "Core Operating Limits Report (COLR)," is being deleted because the Core Operating Limits Report pertains only to an activity that does not apply in a permanently defueled condition. The single safety limit and ten TS LCOs for which limits must be established (listed in TS 5.6.3) are being deleted as described in preceding sections. Therefore, the need for this report no longer exists.

TS 5.6.4, "Post Accident Monitoring Report," is being deleted because the Post-Accident Monitoring Report pertains only to an activity that does not apply in a permanently defueled condition. LCO 3.3.3, which requires this report, is being deleted as described in preceding sections. Therefore, the need for this report no longer exists.

TS 5.6.5, "Steam Generator Tube Inspection Report," is being deleted because the Steam Generator Tube Inspection Report pertains only to an activity that does not apply in a permanently defueled condition. TS 5.5.7, which requires the activity that is the subject of this report, is being deleted as described in preceding sections. Therefore, the need for this report no longer exists.

TS 5.7, High Radiation Area

TS 5.7, "High Radiation Area," provides a description and requirements regarding controls applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR 20 (as provided in paragraph 20.1601(c) of 10 CFR 20). TS 5.7 will remain germane with the reactor permanently defueled. As such, it is being retained as-is with no changes being proposed.

T5.4 SUMMARY

Current Operating Licensed Condition

TS Section 5.0, Administrative Controls, does not contain applicability requirements. As such, all parts of this section can be conservatively defined as being applicable at all times.

Permanently Defueled Condition

Portions of all TS in Section 5.0 will remain germane with the reactor permanently defueled. As such, they are being retained and revised to reflect a permanently defueled condition.

Conclusion

Retaining all TS in Section 5.0, as revised, ensures appropriate requirements for administrative controls.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would revise the Operating License and associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the permanent cessation of reactor operation.

On February 25, 2013, DEK submitted a certification of permanent cessation of power operations pursuant to 10 CFR 50.82(a)(1)(i), stating that DEK had decided to permanently cease power operation of KPS on May 7, 2013. On February 26, 2013, DEK submitted both a Post-Shutdown Decommissioning Activities Report (PSDAR) and an updated Irradiated Fuel Management Report. With the docketing of the subsequent certification for permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii) on May 14, 2013, the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). In support of this condition, the license and associated TS are being proposed for revision to comport to this permanently shutdown and defueled condition in accordance with 10 CFR 50.36(c)(6).

The existing KPS 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. Because the KPS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision, into the PDTS, to provide an acceptable level of safety derived from the reduced scope of postulated design basis accidents associated with a defueled plant, as described in the KPS safety analyses.

DEK has evaluated the proposed amendment to determine if a significant hazards consideration is involved 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

   KPS has permanently ceased operation. The proposed amendment would modify the KPS renewed facility operating license and TS by deleting the portions of the license and TS that are no longer applicable to a permanently defueled facility, while modifying the remaining portions to correspond to the permanently shutdown condition. This change is consistent with the Standard TS and with the criteria set forth in 10 CFR 50.36 for the contents of TS.

Section 14 of the KPS Updated Safety Analysis Report (USAR) described the design basis accident (DBA) and transient scenarios applicable to KPS during power operations. With the reactor in a permanently defueled condition, the spent fuel pool and its systems have been isolated and are dedicated only to spent fuel storage. In this condition the spectrum of credible accidents is much smaller than for an operational plant. As a result of the certifications submitted by DEK 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 in accordance with 10 CFR 50.82(a)(2), most of the accident scenarios postulated in the USAR are no longer possible.

The definition of safety-related structures, systems, and components (SSCs) in 10 CFR 50.2 states that safety-related SSCs are those relied on to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant boundary;
2. The capability to shutdown 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 10 CFR 50.43(a)(1) or 100.11.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after the termination of reactor operations at KPS and the permanent removal of the fuel from the reactor vessel (following 90 days of decay time after shutdown) and purging of the contents of the waste gas decay tanks and liquid waste tanks, none of the SSCs at KPS are required to be relied on for accident mitigation. Therefore, none of the SSCs at KPS meet the definition of a safety-related SSC stated in 10 CFR 50.2 (with the exception of the passive spent fuel pool structure).

The deletion of TS definitions and rules of usage and application, that are currently not applicable in a defueled condition, has no impact on facility 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 KPS has no impact on the remaining DBA (the fuel handling accident in the auxiliary building). The removal of limiting conditions for operation (LCOs) or surveillance requirements (SRs) 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 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 KPS 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 are no longer possible at KPS.

Since KPS has permanently ceased operation, the future generation of fission products has ceased and the remaining source term will decay. The radioactive decay of the irradiated fuel since shutdown of the reactor will have reduced the consequences of the fuel handling accident to levels well below those previously analyzed. The relevant parameter (water level) associated with the fuel pool provides an initial condition for the fuel handling accident analysis and is included in the permanently defueled TS.

The spent fuel pool water level, spent fuel pool boron concentration, and spent fuel pool storage LCOs are retained to preserve the current requirements for safe storage of irradiated fuel.

Fuel pool cooling and makeup related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since there is 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 spent fuel pool.

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 is the only operation currently allowed, and therefore bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation is no longer 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 have no impact on facility SSCs affecting the safe storage of irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of irradiated fuel itself. These changes are consistent with the standard TS. 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 is permanently shutdown and defueled and KPS is no longer authorized to operate the reactor.

The proposed deletion of requirements of the KPS TS do not affect systems credited in the accident analysis for the fuel handling accident in the auxiliary building at KPS. The proposed permanently defueled TS (PDTS) continue to require proper control and monitoring of safety significant parameters and activities.

The proposed restriction on the fuel pool level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA. The spent fuel pool water level, spent fuel pool boron concentration, and spent fuel pool storage LCOs are retained to preserve the current requirements for safe storage of irradiated fuel.

The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (i.e., fuel cladding and spent fuel cooling). Since extended operation in a defueled condition is the only operation currently 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 amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No

Because the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. The only remaining credible accident is a fuel handling accident (FHA). The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses that impact a FHA.

The proposed changes are limited to those portions of TS and license that are not related to the safe storage of irradiated fuel. The requirements for SSCs that have been deleted from the KPS 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 DBAs involving the reactor are no longer possible because the reactor is permanently shutdown and defueled and KPS is no longer authorized to operate the reactor.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety because the current design limits continue to be met for the accident of concern.

Based on the above, Dominion Energy Kewaunee, Inc. concludes that the proposed amendment presents no 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.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the Kewaunee Power Station (KPS) on July 24, 1972 with supplements dated December 18, 1972 and May 10, 1973. The SE, Section 3.1, "Conformance with AEC General Design Criteria," described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

"The Kewaunee plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria."

These General Design Criteria (GDC) are discussed in detail in the KPS USAR.

Design Basis Accidents (DBAs)

Chapter 14 of the KPS USAR described the safety analysis aspects of the plant that were evaluated to demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed the guidelines of 10 CFR 50.67. The analyzed accidents were based on the four categories of plant conditions classified by the American Nuclear Society (ANS) in accordance with the anticipated frequency of occurrence and potential radiological consequences to the public (Normal Operation, Incidents of Moderate Frequency, Infrequent Incidents, and Limiting Faults).

USAR Chapter 14 is divided into three subsections, dealing with various behavior categories. Design basis safety analysis transients are described in Section 14.1. Design basis safety analysis accidents are described in Section 14.2 and Section 14.3.

- Core and Coolant Boundary Protection Analysis, USAR Section 14.1

  The abnormalities presented in Section 14.1 have no off-site radiation consequences. In these events the reactor control and protection system and engineered safeguards are relied upon to protect the core and RCS boundary from damage.

- Standby Safety Features Analysis, USAR Section 14.2

  The accidents presented in Section 14.2 are more severe than those discussed in Section 14.1 and may cause release of radioactive material to the environment. Adequate provisions have been included in the design of the plant, and its standby engineered safeguards to limit potential exposure of the public and the Control Room operator to below the guidelines of 10 CFR 50.67 for these accidents, which could conceivably involve uncontrolled releases of radioactive materials to the environment.

- Reactor Coolant System Pipe Ruptures (LOCA), USAR Section 14.3

  The accident presented in Section 14.3, the rupture of a reactor coolant pipe, is the worst-case accident analyzed and is the primary basis for the design of engineered safety features. It is shown that the consequences of even this accident are within the guidelines of 10 CFR 50.67.

Safety analyses are analyses performed to satisfy regulatory requirements. The safety analyses are integral to the plant's design and licensing basis. The safety analyses demonstrate the integrity of the fission product barriers, the capability to shutdown the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents and transients. Systems, structures, and components (SSC's) that perform design basis functions are credited in the safety analyses for the purpose of mitigating the transient or accident.

Chapter 14 of the KPS USAR described the DBA scenarios that were applicable during plant operations. However, the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). Therefore, most of the accident scenarios postulated in USAR Chapter 14 are no longer applicable. These postulated accidents are listed in the following table (including the USAR section they are described in), with a statement whether or not they are applicable in the permanently defueled condition.

USAR Chapter 14 Postulated Accident or Transient	**Applicability in Defueled Condition**
Core and Coolant Boundary Protection Analysis	
Uncontrolled RCCA Withdrawal from a Sub-critical Condition (14.1.1)	no
Uncontrolled RCCA Withdrawal at Power (14.1.2)	no
RCCA Misalignment (14.1.3)	no
Chemical and Volume Control System Malfunction (14.1.4)	no
Startup of an Inactive Reactor Coolant Loop (14.1.5)	no
Excessive Heat Removal Due to Feedwater System Malfunctions (14.1.6)	no
Excessive Load Increase Incident (14.1.7)	no
Loss of Reactor Coolant Flow (14.1.8)	no
Loss of External Electrical Load (14.1.9)	no
Loss of Normal Feedwater (14.1.10, 14.1.11)	no
Anticipated Transients Without Scram (14.1.12)	no
Loss of all AC Power to the Plant Auxiliaries (14.1.13)	no
Standby Safety Features Analysis	
Fuel Handling Accident (FHA) (14.2.1)	YES
Accidental Release-Recycle of Waste Liquid (14.2.2)	no*
Accidental Release-Waste Gas (14.2.3) - Gas Decay Tank (GDT) Rupture - Volume Control Tank (VCT) Rupture	no*
Steam generator tube rupture (SGTR) (14.2.4)	no
Main Steam Line Break (MSLB) (14.2.5)	no
Rod Ejection (RE) (14.2.6)	no
Locked Reactor Coolant Pump (RCP) Rotor (14.1.8)	no
Large-Break Loss-of-Coolant Accident (LOCA) (14.3.6)	no
Reactor Coolant System Pipe Ruptures (LOCA) (14.3)	no

* With waste gas decay tanks, volume control tanks, liquid holdup tanks, reactor coolant drain tank, and associated systems purged of their contents.

As discussed in USAR Section 14.2.2, the potential hazard associated with accidental waste liquid releases is derived only from the volatilized components. With the associated lines and tanks drained, this hazard no longer exists. Likewise, waste gas release from a tank that has been purged of its contents is no longer possible.

USAR Section 14.2.1 describes the fuel handling accident. Since KPS is no longer authorized to operate or to place or retain fuel in the reactor (following docketing of the certification specified in 10 CFR 50.82(a)(2)), a fuel handling accident onto the top of the core (or elsewhere within containment) is no longer possible and therefore no longer part of the licensing basis. However, a fuel handling accident in the auxiliary building (including the spent fuel pool) is still possible and is analyzed.

10 CFR 50.2, Definitions, Safety-Related Structures, Systems and Components

10 CFR 50.2 defines safety-related structures, systems, and components (SSCs) as 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.

10 CFR 50.36, Technical Specifications

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of Technical Specifications (TS). 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 TS "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.

In September 1992, the Commission issued NUREG-1431, which was developed using the guidance and criteria contained in the Commission's Interim Policy Statement. Standard Technical Specifications (STS) were established as a model for developing improved TS for Westinghouse plants in general. STS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which was published in a "Split Report" issued to the Nuclear Steam System Supplier (NSSS) Owners Groups in May 1988. STS also reflect the results of extensive discussions concerning various drafts of STS, so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all NSSS designs. As such, the generic Bases presented in NUREG-1431 provide an abundance of information regarding the extent to which the STS present requirements that are necessary to protect public health and safety.

On July 22, 1993, the Commission issued its Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, indicating that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the safety benefits of the improved STS, and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments, and for complete conversions to improved STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated surveillances should remain in the 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 Technical Specification limiting conditions for operation 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 is present in the reactor or reactor coolant system at the KPS facility following permanent defueling, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that Technical Specification limiting conditions for operation must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident [DBA] 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 design basis accident and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some design basis accidents which continue to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The DBAs for KPS are discussed within this proposed amendment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that Technical Specification limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident 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 Technical Specifications 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 design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that continue to apply to KPS, there is still a design basis accident that continues to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The scope of DBAs applicable to KPS is discussed in more detail within this proposed amendment.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that Technical Specification limiting conditions for operation 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 Technical Specification limiting conditions for operation. All of the accident sequences that previously dominated risk at KPS are no longer applicable with the reactor in the permanently shutdown and defueled condition.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they "...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, therefore, are the provisions that the Commission deems essential for the safe operation of the facility that are not already covered by other regulations. Accordingly, the NRC staff determined that administrative control requirements that are not specifically required under Section 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or an event giving rise to an immediate threat to the public health and safety, may be relocated to more appropriate documents (e.g., Quality Assurance Program, Security Plan, or Emergency Plan), which are subject to regulatory controls. Similarly, while the required content of TS administrative controls is specified in 10 CFR 50.36(c)(5), particular details may be relocated to licensee-controlled documents, where other regulations provide adequate regulatory control.

10 CFR 50.36(c)(6), "Decommissioning," applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1). For such facilities, 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.

On February 2, 2011, the NRC issued Amendment No. 207 to Facility Operating License No. DPR-43 for KPS. The amendment consisted of changes to the Technical Specifications and the license conditions for KPS. As stated in the NRC safety evaluation accompanying Amendment 207, the amendment converted the previous custom TS (CTS) to the improved TS (ITS) and relocated certain requirements to other licensee-controlled documents. The KPS ITS are based on:

- NUREG-1431, "Standard Technical Specifications (STS) Westinghouse Plants," Revision 3.0;
- "NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," dated July 22, 1993 (58 FR 39132); and
- 10 CFR 50.36, "Technical Specifications."

This proposed amendment deletes the portions of the previous KPS TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shutdown condition, consistent with STS.

10 CFR 50.51, Continuation of License

10 CFR 50.51(b) states "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."

10 CFR 50.82, Termination of License

10 CFR 50.82(a)(2) states "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."

5.3 Precedent

This proposed amendment is consistent with the existing license currently in effect for Millstone Nuclear Power Station (DPR-21), which was last substantively revised on March 31, 2001 (Reference 5). The Millstone license amendment that was issued to reflect the permanently shutdown status of the plant on November 9, 1999 (Reference 6), contains license conditions and Technical Specifications (TS) similar to those being proposed herein.

This proposed amendment is also consistent with the license, and accompanying TS, issued to Zion Nuclear Power Station on December 30, 1999 (Reference 7), which was issued to reflect the permanently shutdown status of the plant.

5.4 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.

6.0 ENVIRONMENTAL CONSIDERATION

DEK has evaluated this license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. DEK has determined that this license amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement.

However, (i) the proposed amendment involves no significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite, and (iii) there is no 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).

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

7.0 REFERENCES

1. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Certification of Permanent Cessation of Power Operations," dated February 25, 2013 (ADAMS Accession No. ML13058A065).
2. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Post-Shutdown Decommissioning Activities Report," dated February 26, 2013 (ADAMS Accession No. ML13063A248).
3. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Update to Irradiated Fuel Management Plan Pursuant to 10 CFR 50.54(bb)," dated February 26, 2013 (ADAMS Accession No. ML13059A028).
4. NRC Safety Evaluation Report Related to the License Renewal of Kewaunee Power Station (TAC No. MD9408), dated November 4, 2010 (ADAMS Accession No. ML103000131).
5. Millstone Nuclear Power Station, Unit 1, Amendment No.109, License No. DPR-21, Date of Issuance March 31, 2001 (ADAMS Accession No. ML010920303).
6. NRC Safety Evaluation for Millstone Power Station Unit 1 in License Amendment 106 to DPR-21, dated November 9, 1999 (ADAMS Accession Nos. ML993330283 and ML993330269).
7. NRC Safety Evaluation for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively (License Nos. DPR-39 and DPR-48)), dated December 30, 1999 (ADAMS Accession Nos. ML003672704 and ML003672696).
8. Letter from Eugene S. Grecheck (DEK) to NRC Document Control Desk, "License Amendment Request 255: Deletion of License Renewal Condition for Permanently Defueled License," dated April 16, 2013.
9. Letter from Daniel G. Stoddard (DEK) to NRC Document Control Desk, "Certification of Permanent Removal of Fuel from the Reactor Vessel," dated May 14, 2013 (ADAMS Accession No. ML13135A209).

ATTACHMENT 2

LICENSE AMENDMENT REQUEST 256
PERMANENTLY DEFUELED LICENSE AND TECHNICAL SPECIFICATIONS

MARKED UP TECHNICAL SPECIFICATIONS PAGES:

TS	**"TABLE OF CONTENTS"**
TS 1.1	**"Definitions"**
TS 1.2	**"Logical Connectors"** (not attached; retain unchanged)
TS 1.3	**"Completion Times"**
TS 1.4	**"Frequency"**
TS 3.0	**"Limiting Condition for Operation (LCO) Applicability"**
TS 3.0	**"Surveillance Requirement (SR) Applicability"**
TS 3.7.13	**"Spent Fuel Pool Water Level"**
TS 3.7.14	**"Spent Fuel Pool Boron Concentration"**
TS 3.7.15	**"Spent Fuel Pool Storage"**
TS 4.1	**"Site Location"**
TS 4.3	**"Fuel Storage"**
TS 5.1	**"Responsibility"**
TS 5.2	**"Organization"**
TS 5.3	**"Unit Staff Qualifications"**
TS 5.4	**"Procedures"**
TS 5.5	**"Programs and Manuals"**
TS 5.6	**"Reporting Requirements"**
TS 5.7	**"High Radiation Area"**

(TS not listed above are deleted in their entirety.)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

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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.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
CERTIFIED FUEL HANDLER ~~AXIAL FLUX DIFFERENCE (AFD)~~	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2~~AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector~~.
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. 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.

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT 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 COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
~~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 parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. 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 per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using ICRP-30, 1979, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~
~~DOSE EQUIVALENT XE-133~~	~~DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."~~

1.1 Definitions

~~LEAKAGE~~

~~LEAKAGE shall be:~~

~~a. Identified LEAKAGE~~

~~1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,~~

~~2. LEAKAGE into the containment 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, or~~

~~3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);~~

~~b. Unidentified LEAKAGE~~

~~All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and~~

~~c. Pressure Boundary LEAKAGE~~

~~LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.~~

~~MODE~~

~~A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.~~

OPERABLE – OPERABILITY

A system, subsystem, train, 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, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

1.1 Definitions

~~PHYSICS TESTS~~	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:~~ ~~a. Described in Chapter 13, "Initial Test and Operation," of the USAR;~~ ~~b. Authorized under the provisions of 10 CFR 50.59; or~~ ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~
~~QUADRANT POWER TILT RATIO (QPTR)~~	~~QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.~~
~~RATED THERMAL POWER (RTP)~~	~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1772 MWt.~~
~~SHUTDOWN MARGIN (SDM)~~	~~SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:~~ ~~a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and~~ ~~b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power level design temperature.~~
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.

1.1 Definitions

~~THERMAL POWER~~	~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

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

~~MODE~~	~~TITLE~~	~~REACTIVITY CONDITION (k_{eff})~~	~~% RATED THERMAL POWER[a]~~	~~AVERAGE REACTOR COOLANT TEMPERATURE (°F)~~
~~1~~	~~Power Operation~~	~~≥ 0.99~~	~~> 5~~	~~NA~~
~~2~~	~~Startup~~	~~≥ 0.99~~	~~≤ 5~~	~~NA~~
~~3~~	~~Hot Standby~~	~~< 0.99~~	~~NA~~	~~≥ 350~~
~~4~~	~~Hot Shutdown[b]~~	~~< 0.99~~	~~NA~~	~~$350 > T_{avg} > 200$~~
~~5~~	~~Cold Shutdown[b]~~	~~< 0.99~~	~~NA~~	~~≤ 200~~
~~6~~	~~Refueling[c]~~	~~NA~~	~~NA~~	~~NA~~

~~(a) Excluding decay heat.~~

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

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

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~~management of irradiated 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 ~~unit~~ facility 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 ~~unit~~ facility 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 trains, 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.

However, when a subsequent train, 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.

1.3 Completion Times

DESCRIPTION (continued)

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 extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, 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 . . ."

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~~Verify.... AND B.2 ~~Be in MODE 5~~Restore....	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

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~~perform the verification required by ACTION B.1 within 6 hours AND perform the restoration required by ACTION B.2 ~~in MODE 5~~ within 36 hours. A total of 6 hours is allowed for performing ACTION B.1 ~~reaching MODE 3~~ and a total of 36 hours (not 42 hours) is allowed for performing ACTION B.2 ~~reaching MODE 5~~ from the time that Condition B was entered. If ACTION B.1~~MODE 3~~ is ~~reached~~ completed within 3 hours, the time allowed for ~~reaching MODE 5~~completing ACTION B.2 is the next 33 hours because the total time allowed for completing ACTION B.2 ~~reaching MODE 5~~ is 36 hours.

If Condition B is entered while performing the verification required by ACTION B.1~~in MODE 3~~, the time allowed for completing ACTION B.2 ~~reaching MODE 5~~ is the next 36 hours.

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~~Verify.... AND B.2 ~~Be in MODE 5~~Initiate....	6 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.

1.3 Completion Times

~~EXAMPLES (continued)~~

~~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.~~

~~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.~~

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

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

1.3 Completion Times

EXAMPLES (continued)

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.

It is 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. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

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~~Verify.... AND B.2 ~~Be in MODE 4~~Initiate....	6 hours 12 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.

1.3 Completion Times

EXAMPLES (continued)

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 (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

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~~Verify.... AND B.2 ~~Be in MODE 4~~Initiate....	6 hours 12 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.

1.3 Completion Times

EXAMPLES (continued)

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.

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. OR A.2 ~~Reduce THERMAL POWER to ≤ 50% RTP~~Verify....	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 ~~Be in MODE 3~~Initiate....	6 hours

1.3 Completion Times

EXAMPLES (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. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. 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.

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated. AND A.2 Restore subsystem to OPERABLE status.	1 hour AND Once per 8 hours thereafter 72 hours
B. Required Action and associated Completion Time not met.	B.1 ~~Be in MODE 3~~Verify.... AND B.2 ~~Be in MODE 5~~Initiate....	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

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 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 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.

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 specific 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 criteria.

~~Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:~~

1.4 Frequency

~~DESCRIPTION (continued)~~

~~a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or~~

~~b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or~~

~~c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.~~

~~Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.~~

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is the applicable specified condition~~MODES 1, 2, and 3~~.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

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 stated 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 unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit 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 Example 1.4-3), then SR 3.0.3 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a ~~MODE or other~~ specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the ~~MODE or other~~ specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ~~≥ 25% RTP~~entry into the applicable condition 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 the applicable condition is entered~~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 25% extension allowed by SR 3.0.2. "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 applicable condition is exited, the measurement of both intervals stops. New intervals start upon ~~reactor power reaching 25% RTP~~entering the applicable condition.

1.4 Frequency

~~EXAMPLES (continued)~~

~~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 performed within the "specified Frequency." Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, 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.~~

~~Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance was 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.~~

1.4 Frequency

~~EXAMPLES (continued)~~

~~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 was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), 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 was not met), SR 3.0.4 would require satisfying the SR.~~

1.4 Frequency

~~EXAMPLES (continued)~~

~~EXAMPLE 1.4-5~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~NOTE~~ ~~Only required to be performed in MODE 1.~~ ~~Perform complete cycle of the valve.~~	~~7 days~~

~~The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.~~

~~As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, 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 result in entry into MODE 1.~~

~~Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance was not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.~~

1.4 Frequency

~~EXAMPLES (continued)~~

~~EXAMPLE 1.4-6~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~NOTE~~ ~~Not required to be met in MODE 3.~~ ~~Verify parameter is within limits.~~	~~24 hours~~

~~Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). 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 was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, 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 to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency was 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~~, LCO 3.0.7, and LCO 3.0.8~~.
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	Deleted~~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 3 within 7 hours;~~ ~~b. MODE 4 within 13 hours; and~~ ~~c. MODE 5 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, 3, and 4.~~
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; 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.

3.0 LCO Applicability

~~LCO 3.0.4 (continued)~~

This Specification 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~~.

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, an evaluation shall be performed in accordance with Specification 5.5.13, "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.

~~LCO 3.0.7~~

~~Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.~~

3.0 LCO Applicability

~~LCO 3.0.8~~

~~When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:~~

~~a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or~~

~~b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 24 hours.~~

~~At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.~~

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. 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.

3.0 SR APPLICABILITY

~~SR 3.0.4 (continued)~~

~~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.~~

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Water Level

LCO 3.7.13 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent 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 spent fuel pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO 3.7.14 The spent fuel pool boron concentration shall be ≥ 240 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	~~NOTE~~ ~~LCO 3.0.3 is not applicable.~~	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	<u>OR</u>	
	A.2.2 Initiate action to perform a spent fuel pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each fuel assembly stored in the spent fuel pool shall be in accordance with the following:

a. Irradiated fuel assemblies discharged prior to or during the 1984 refueling outage with a combination of burnup and initial nominal enrichment in the "Acceptable Domain" of Figure 3.7.15-1 shall be stored in the transfer canal spent fuel pool or the north and south combined spent fuel pools; and

b. ~~New fuel assemblies, i~~Irradiated fuel assemblies discharged after the 1984 refueling outage, and irradiated fuel assemblies discharged prior to or during the 1984 refueling outage with a combination of burnup and initial nominal enrichment in the "Unacceptable Domain" of Figure 3.7.15-1, shall be stored in the north and south combined spent fuel pools.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	~~------------NOTE------------~~ ~~LCO 3.0.3 is not applicable.~~ Initiate action to move the noncomplying fuel assembly to an acceptable location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1.	Prior to storing the fuel assembly in the spent fuel pool

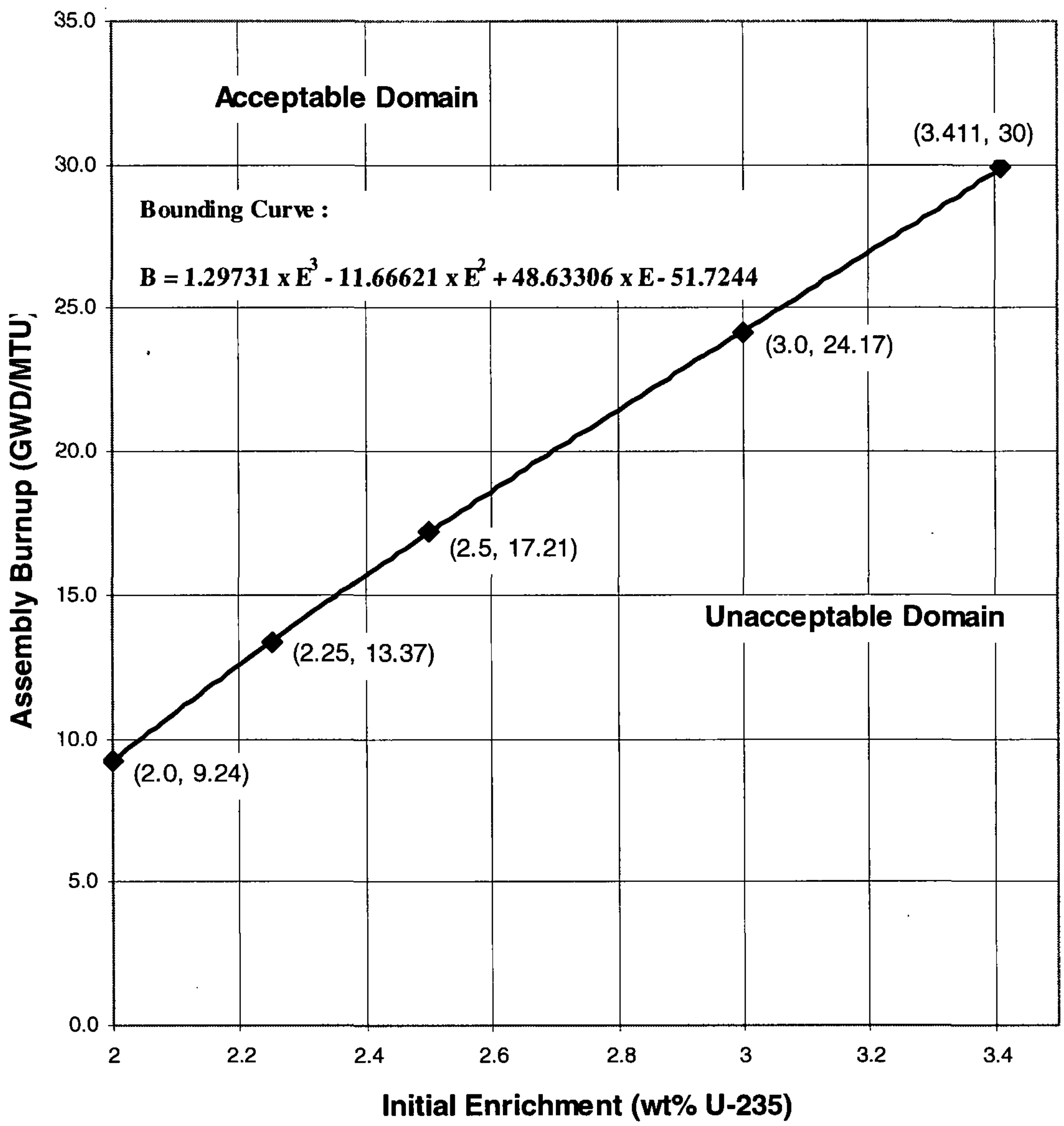


Figure 3.7.15-1 (page 1 of 1)
Fuel Assembly Burnup Limits in the Spent Fuel Pools

4.0 DESIGN FEATURES

4.1 Site Location

The Kewaunee Power Station is located on property owned by Dominion Energy Kewaunee Inc. at a site on the west shore of Lake Michigan, approximately 30 miles east-southeast of the city of Green Bay, Wisconsin.

The minimum distance from the center line of the reactor containment to the site exclusion radius as defined in 10 CFR 100.3 is 1200 meters.

4.2 ~~Reactor Core~~Deleted

~~4.2.1 Fuel Assemblies~~

~~The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. 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 fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.~~

~~4.2.2 Control Rod Assemblies~~

~~The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.~~

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. Fuel assemblies having a maximum U-235 enrichment of 4.9776 weight percent;

b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR;

c. A nominal 8.3 inch rack cell lattice spacing between fuel assemblies placed in the canal pool;

4.3 Fuel Storage (continued)

d. A minimum 10 inch center to center distance between fuel assemblies placed in the north and south pools; and

e. New and spent fuel assemblies stored in the north and south pools and the canal pool in accordance with LCO 3.7.15, "Spent Fuel Pool Storage."

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

~~a. Fuel assemblies having a maximum U-235 enrichment of 4.9776 weight percent;~~

~~b. k_{eff} < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR;~~

~~c. k_{eff} < 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.5 of the USAR; and~~

~~d. A nominal 21 inch center to center distance between fuel assemblies placed in the 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 645 ft 2 inches (mean sea level).

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1205 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall ~~unit~~ plant 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 or modification to systems or equipment that affect nuclear safety.

5.1.2 The shift manager shall be responsible for the ~~control room~~shift command function. ~~During any absence of the shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Operator license shall be designated to assume the control room command function. During any absence of the shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active Senior Operator license or Operator license shall be designated to assume the control room command function.~~

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

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

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 shall be documented in the quality assurance program. The plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be maintained in appropriate plant documents.

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 storage ~~operation~~ and maintenance of the nuclear fuel~~plant~~.

c. A specified corporate officer 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 safe management of nuclear ~~safety~~fuel.

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

5.2.2 ~~Unit~~ Facility Staff

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

a. ~~A non-licensed operator shall be assigned if the reactor contains fuel and an additional non-licensed operator shall be assigned if the reactor is operating in MODES 1, 2, 3, or 4.~~ Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

b. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1~~10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e~~ for a period of time not to exceed 2 hours, except in severe weather conditions, in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5.2 Organization

5.2.2 ~~Unit~~ Facility Staff (continued)

c. ~~A radiation technologist shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, except in severe weather conditions, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.~~ All fuel handling operations shall be directly supervised by a qualified individual.

d. The ~~operations~~ shift manager ~~or assistant operations manager~~ shall ~~hold~~ be a CERTIFIED FUEL HANDLER~~Senior Operator license~~.

~~e. When the unit is in MODE 1, 2, 3, or 4 an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition

POSITION	MINIMUM STAFFING
CERTIFIED FUEL HANDLER	1
Non-Certified Operator	1

Note: The Non-Certified Operator position may be filled by a CERTIFIED FUEL HANDLER.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility~~Unit~~ Staff Qualifications

5.3.1 Each member of the ~~unit~~ facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for:

a. The radiation protection manager who shall meet or exceed the recommendation of Regulatory Guide 1.8, Revision 1-R, September 1975, or their equivalent as further clarified in Attachment 1 to the NRC Safety Evaluation Report enclosed with Amendment No. 46, dated July 12, 1982.

~~b. The education and experience eligibility requirements for operator license applicants, changes thereto, shall be those previously reviewed and approved by the NRC, specifically those referenced in NRC Safety Evaluation letter for Amendment 170, dated October 2, 2003.~~

5.3.2 An NRC approved training and retraining program for the CERTIFIED FUEL HANDLERS shall be maintained~~For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m)~~.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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;

b. ~~The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33~~Deleted;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and

    b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. 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 Programs and Manuals (continued)

5.5.2 Deleted ~~Primary Coolant Sources Outside Containment~~

~~The System Integrity Program (SIP) 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 Safety Injection System, Chemical and Volume Control System, Containment Spray System, Miscellaneous Sumps and Drains System, Reactor Building Ventilation System, Residual Heat Removal System, and Primary Sampling System. The program shall include the following:~~

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

~~b. Integrated leak test requirements for each system at least once per 18 months.~~

~~The provisions of SR 3.0.2 are applicable.~~

5.5.3 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 Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

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 each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

5.5 Programs and Manuals

5.5.3 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 from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin; and

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

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit 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 each unit 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, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies.

5.5 Programs and Manuals (continued)

5.5.4 Deleted~~Component Cyclic or Transient Limit Program~~

~~This program provides controls to track the USAR, Section 4, cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

5.5.5 Deleted~~Reactor Coolant Pump Flywheel Inspection Program~~

~~This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~

~~In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.~~

5.5.6 Deleted~~Inservice Testing Program~~

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.~~

~~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~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 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.~~

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

5.5 Programs and Manuals (continued)

5.5.7 Deleted~~Steam Generator (SG) Program~~

~~A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:~~

~~a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.~~

~~b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.~~

~~1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day per SG.~~

~~3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."~~

5.5 Programs and Manuals

~~5.5.7 Steam Generator (SG) Program (continued)~~

~~c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.~~

~~d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

~~1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.~~

~~2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

~~3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.~~

~~e. Provisions for monitoring operational primary to secondary LEAKAGE.~~

5.5 Programs and Manuals (continued)

5.5.8 Deleted~~Secondary Water Chemistry Program~~

~~This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:~~

~~a. Identification of a sampling schedule for the critical variables and control points for these variables;~~

~~b. Identification of the procedures used to measure the values of the critical variables;~~

~~c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;~~

~~d. Procedures for the recording and management of data;~~

~~e. Procedures defining corrective actions for all off control point chemistry conditions; and~~

~~f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.~~

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

~~A program shall be established to implement the following required testing of safety related filter ventilation systems in accordance with Regulatory Positions C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, ANSI N510-1975, ATSM D3803-1989, and AG-1.~~

~~The test described in Specification 5.5.9.a shall be performed once per 18 months and after each complete or partial replacement of the high efficiency particulate air (HEPA) filter bank and any maintenance on the system that could affect the HEPA bank bypass leakage.~~

~~The test described in Specification 5.5.9.b shall be performed after each complete or partial replacement of a charcoal adsorber bank or maintenance on the system that could affect the charcoal adsorber bank bypass leakage.~~

~~The test described in Specification 5.5.9.c shall be performed once per 18 months for filters in a standby status or after 720 hours of filter operation, and following painting, fire, or chemical release in any ventilation zone communicating with the system.~~

~~The test described in Specification 5.5.9.d shall be performed once per 18 months.~~

5.5 Programs and Manuals

~~5.5.9 Ventilation Filter Testing Program (VFTP) (continued)~~

~~The test described in Specification 5.5.9.e shall be performed after any maintenance or testing that could affect the air distribution within the systems.~~

~~a. Demonstrate for each of the safety related systems listed below that an inplace test of the HEPA filters shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Position C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%.~~

~~Safety Related System~~	~~Flow Rate (cfm)~~
~~Shield Building Ventilation System (SBVS)~~	~~5700~~
~~Auxiliary Building Special Ventilation (ASV) System~~	~~9000~~
~~Control Room Post Accident Recirculation (CRPAR) System~~	~~2500~~

~~b. Demonstrate for each of the safety related systems listed below that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Position C.5.d of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%.~~

~~Safety Related System~~	~~Flow Rate (cfm)~~
~~SBVS~~	~~5700~~
~~ASV System~~	~~9000~~
~~CRPAR System~~	~~2500~~

~~c. Demonstrate for each of the safety related systems listed below that a laboratory test of a sample of the charcoal adsorber shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 and AG-1 at a temperature of 30°C (86°F) and relative humidity of 95%.~~

~~Safety Related System~~	~~Penetration~~
~~SBVS~~	~~≤ 2.5%~~
~~ASV System~~	~~≤ 2.5%~~
~~CRPAR System~~	~~≤ 5%~~

5.5 Programs and Manuals

~~5.5.9 Ventilation Filter Testing Program (VFTP) (continued)~~

~~d. Demonstrate for each of the safety related systems listed below that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1975 at the system flowrate specified below ± 10%.~~

~~Safety Related System~~	~~Combined Delta P (in. wc)~~	~~Flow Rate (cfm)~~
~~SBVS~~	~~< 6.3~~	~~5700~~
~~ASV System~~	~~< 6.3~~	~~9000~~
~~CRPAR System~~	~~< 2.4~~	~~2500~~

~~e. Demonstrate for each of the safety related systems listed below that when tested at the system flowrate specified below (± 10%) the air distribution is uniform within ± 20%.~~

~~Safety Related System~~	~~Flow Rate (cfm)~~
~~SBVS~~	~~5700~~
~~ASV System~~	~~9000~~

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

5.5.10 Deleted~~Explosive Gas and Storage Tank Radioactivity Monitoring Program~~

~~This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radioactive Waste Disposal System, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."~~

~~The program shall include:~~

~~a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radioactive Waste Disposal 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);~~

5.5 Programs and Manuals

~~5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)~~

~~b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tank's contents; and~~

~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Waste Disposal System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

~~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.~~

5.5.11 Deleted~~Diesel Fuel Oil Testing Program~~

~~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 flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and~~

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

~~b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and~~

~~c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 92 days.~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.~~

5.5 Programs and Manuals (continued)

5.5.12 Technical Specifications (TS) Bases Control Program

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

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

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

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

2. A change to the updated 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 Specification 5.5.12.b 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).

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

~~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 actions may 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.~~

~~a. The SFDP shall contain the following:~~

~~1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;~~

~~2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;~~

~~3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and~~

~~4. Other appropriate limitations and remedial or compensatory actions.~~

5.5 Programs and Manuals

~~5.5.13 Safety Function Determination Program (SFDP) (continued)~~

~~b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), 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~~

~~1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or~~

~~2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or~~

~~3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.13.b.1 and 5.5.13.b.2 above is also inoperable.~~

~~c. 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.~~

5.5.14 Deleted~~Containment Leakage Rate Testing Program~~

~~a. A program shall establish the leakage rate testing of the 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.~~

~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident,~~ $\cancel{P_a}$, ~~is 44.6 psig. The containment design pressure is 46 psig.~~

~~c. The maximum allowable containment leakage rate,~~ $\cancel{L_a}$, ~~at 46 psig (Peak Test Pressure), shall be 0.2% of containment air weight per day.~~

~~d. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is 1.0~~ $\cancel{L_a}$. ~~During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are~~ $\cancel{< 0.60 L_a}$ ~~for the Type B and C tests and~~ $\cancel{\leq 0.75 L_a}$ ~~for Type A tests.~~

5.5 Programs and Manuals

~~5.5.14 Containment Leakage Rate Testing Program (continued)~~

~~2. Air lock door seal leakage testing acceptance criteria for each door seal is a leakage rate of < 0.005 L_a when pressurized to ≥ 10 psig.~~

~~e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing program.~~

5.5.15 Deleted~~Battery Monitoring and Maintenance Program~~

~~This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:~~

~~a. The program allows the following RG 1.129, Revision 2 exceptions:~~

~~1. Battery temperature correction may be performed before or after conducting discharge tests.~~

~~2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.~~

~~3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery after each performance of SR 3.8.6.5."~~

~~4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."~~

~~5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration", the following may be used: "Following the test, record the float voltage of each cell of the string."~~

~~b. The program shall include the following provisions:~~

~~1. Actions to restore battery cells with float voltage < 2.13 V;~~

5.5 Programs and Manuals

~~5.5.15 Battery Monitoring and Maintenance Program (continued)~~

~~2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;~~

~~3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;~~

~~4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and~~

~~5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.~~

5.5.16 Deleted~~Setpoint Control Program~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analysis. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verify that instrumentation will function as required.~~

~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation";~~

~~2. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions";~~

~~3. LCO 3.3.5, "Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation";~~

~~4. LCO 3.3.6, "Containment Purge and Vent Isolation Instrumentation"; and~~

~~5. LCO 3.3.7, "Control Room Post Accident Recirculation (CRPAR) Actuation Instrumentation."~~

~~b. The program shall list the value of the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) for each Function described in Paragraph a. The NRC staff has not approved processing changes to Kewaunee Power Station instrumentation setpoints under 10 CFR 50.59 using an approved setpoint methodology as described in Option B of TSTF-493. NRC approval using~~

5.5 Programs and Manuals

~~5.5.16 Setpoint Control Program (continued)~~

~~10 CFR 50.90 is required to change the listed value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in Paragraph a.~~

~~c. The program shall establish methods to ensure that Functions described in Paragraph a. will function as required by verifying the as-left and as-found settings are consistent with the list of values established by Paragraph b. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~d. The program shall identify the Functions described in Paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.~~

~~1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.~~

~~2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~

~~e. The program shall be specified in the Technical Requirements Manual.~~

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 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 15 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.

5.6.2 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the ~~unit~~ facility in the previous year shall be submitted by May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous 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 Part 50, Appendix I, Section IV.B.1.

5.6.3 Deleted ~~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. SL 2.1.1, "Reactor Core SLs";~~

~~2. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";~~

~~3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";~~

~~5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

4. ~~LCO 3.1.5, "Shutdown Bank Insertion Limits";~~

5. ~~LCO 3.1.6, "Control Bank Insertion Limits";~~

6. ~~LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";~~

7. ~~LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";~~

8. ~~LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";~~

9. ~~LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Functions 6 and 7 (Overtemperature ΔT and Overpower ΔT, respectively);~~

10. ~~LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and~~

11. ~~LCO 3.9.1, "Boron Concentration."~~

b. ~~The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% of the original RATED THERMAL POWER is specified in a previously approved method, 100.6% of uprated RATED THERMAL POWER may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow Ultrasonic Flow Measurement System (Crossflow System) as described in the Reference listed in Specification 5.6.3.b.14 below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.~~

~~Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow System is used for main feedwater flow measurement.~~

~~The approved analytical methods are described in the following documents:~~

1. ~~Topical Report WPSRSEM-NP, "Kewaunee Nuclear Power Plant – Review For Kewaunee Reload Safety Evaluation Methods."~~

2. ~~WCAP-12945-P-A (Proprietary), "Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," Volume I and Volume II-V.~~

~~5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

3. ~~WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."~~

4. ~~WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2.~~

5. ~~XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup."~~

6. ~~ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU."~~

7. ~~EMF-92-116 (P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs."~~

8. ~~WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification."~~

9. ~~WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology."~~

10. ~~WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions."~~

11. ~~WCAP-14449-P-A, "Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection."~~

12. ~~WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."~~

13. ~~WCAP-11397-P-A, "Revised Thermal Design Procedure."~~

14. ~~CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology."~~

15. ~~Topical Report DOM-NAF-5-A, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)."~~

~~The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).~~

~~5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~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.~~

5.6.4 Deleted ~~Post Accident Monitoring Report~~

~~When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.~~

5.6.5 Deleted ~~Steam Generator Tube Inspection Report~~

~~A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.7, "Steam Generator (SG) Program." The report shall include:~~

~~a. The scope of inspections performed on each SG;~~

~~b. Active degradation mechanisms found;~~

~~c. Nondestructive examination techniques utilized for each degradation mechanism;~~

~~d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;~~

~~e. Number of tubes plugged during the inspection outage for each active degradation mechanism;~~

~~f. Total number and percentage of tubes plugged to date;~~

~~g. The results of condition monitoring, including the results of tube pulls and in-situ testing; and~~

~~h. The effective plugging percentage for all plugging in each SG.~~

ATTACHMENT 3

LICENSE AMENDMENT REQUEST 256
PERMANENTLY DEFUELED LICENSE AND TECHNICAL SPECIFICATIONS

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES:

TS B 3.0	**"Limiting Condition for Operation (LCO) Applicability"**
TS B 3.0	**"Surveillance Requirement (SR) Applicability"**
TS B 3.7.13	**"Spent Fuel Pool Water Level"**
TS B 3.7.14	**"Spent Fuel Pool Boron Concentration"**
TS B 3.7.15	**"Spent Fuel Pool Storage"**

(TS Bases not listed above are deleted in their entirety.)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.8 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 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~~action may be required to place the ~~unit~~ system 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 ~~unit~~ system 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.

BASES

LCO 3.0.2 (continued)

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 Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. ~~An example of this is in LCO 3.4.3, "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. ~~Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may 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~~a 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.

LCO 3.0.3

Deleted~~LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:~~

~~a. An associated Required Action and Completion Time is not met and no other Condition applies; or~~

~~b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.~~

BASES

~~LCO 3.0.3 (continued)~~

~~This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.~~

~~Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, "Completion Times."~~

~~A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:~~

~~a. The LCO is now met;~~

~~b. A Condition exists for which the Required Actions have now been performed; or~~

~~c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.~~

~~The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching~~

BASES

~~LCO 3.0.3 (continued)~~

~~MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.~~

~~In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.~~

~~Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.13, "Spent Fuel Pool Water Level." LCO 3.7.13 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.13 to "Suspend movement of irradiated fuel assemblies in the spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.~~

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. It allows placing the unit in a ~~MODE or other~~ specified condition stated in that Applicability (i.e., the Applicability desired to be entered) when ~~unit~~ system conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a ~~MODE or other~~ specified condition in the Applicability with the LCO not met 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. Compliance with Required Actions that permit continued operation of the ~~unit~~ system for an unlimited period of time in a ~~MODE or other~~ specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the ~~unit~~ system before or after the ~~MODE change~~specified condition. Therefore, in such cases, entry into a ~~MODE or other~~ specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.b allows entry into a ~~MODE or other~~ specified condition in the Applicability with the LCO not met 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.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed ~~MODE~~ specified condition change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the ~~MODE or other~~ specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

BASES

LCO 3.0.4 (continued)

~~The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, t~~There is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a ~~MODE or other~~ specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into ~~MODES or other~~ specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters ~~(e.g., Containment Air Temperature, Containment Pressure, and Moderator Temperature Coefficient)~~, and may be applied to other Specifications based on NRC plant specific approval.

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

The provisions of LCO 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 LCO 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a entry into MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.~~

BASES

LCO 3.0.4 (continued)

Upon entry into a ~~MODE or other~~ specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

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. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a ~~containment isolation~~ valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the ~~unit~~ system is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the ~~unit~~ system is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, 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.

~~Specification 5.5.13, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.~~

BASES

~~LCO 3.0.6 (continued)~~

~~One aspect of the SFPD is the provision for cross train checks. The SFDP requires the performance of cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. 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 (EXAMPLE B 3.0.6-1);~~

b. ~~A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2); or~~

c. ~~A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).~~

~~EXAMPLE B 3.0.6-1 (Refer to Figure B 3.0-1)~~

~~If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.~~

~~EXAMPLE B 3.0.6-2 (Refer to Figure B 3.0-1)~~

~~If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.~~

~~EXAMPLE B 3.0.6-3 (Refer to Figure B 3.0-1)~~

~~If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10, and 11.~~

~~If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

~~This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross~~

BASES

~~LCO 3.0.6 (continued)~~

~~train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).~~

~~When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.~~

~~LCO 3.0.7~~

~~There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.~~

~~The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.~~

BASES

~~LCO 3.0.8~~

~~LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control.~~

~~If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.~~

~~LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.~~

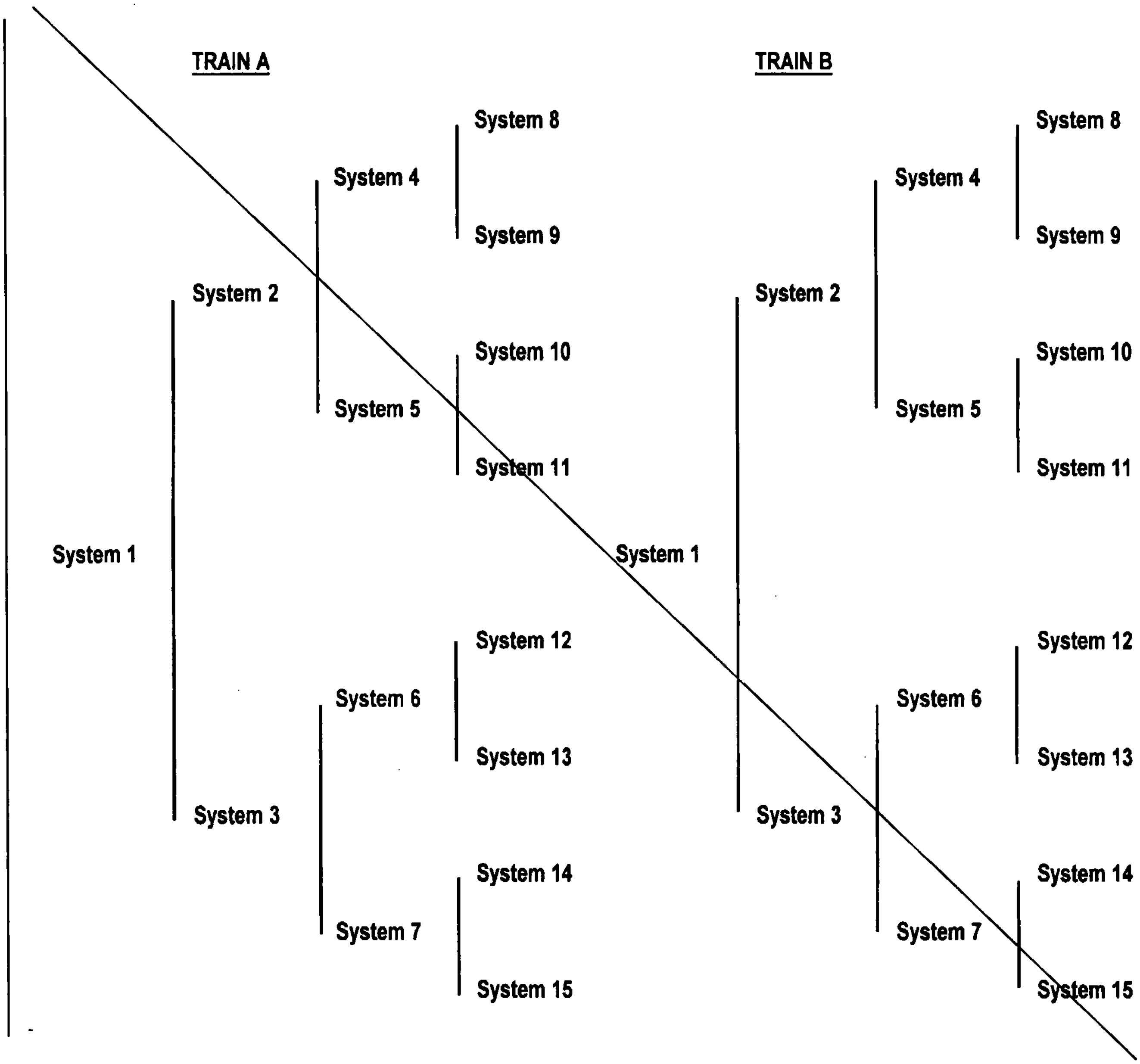
~~When LCO 3.0.8.a is used at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), must be available.~~

~~LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 24 hours to restore the snubber(s) before declaring the supported system inoperable. The 24 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.~~

~~When LCO 3.0.8.b is used at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling (e.g., F&B, firewater system or "aggressive secondary cooldown" using the steam generators) must be available.~~

BASES

~~LCO 3.0.8 (continued)~~	~~LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.~~



~~Figure B 3.0-1~~
~~Configuration of Trains and Systems~~

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. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

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 not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a ~~MODE or other~~ specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. ~~The SRs associated with a Test Exception LCO are only applicable when the Test Exception LCO is used as an allowable exception to the requirements of a Specification.~~

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR.

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.

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.

~~An example of this process is: Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures ≥ 500 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the 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).

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. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the

BASES

SR 3.0.2 continued)

regulations. 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 that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit 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

BASES

SR 3.0.3 (continued)

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~~ system 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, 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 action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should 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 licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

BASES

SR 3.0.3 (continued)

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.

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 a Surveillance not being met in accordance with LCO 3.0.4.

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.

BASES

SR 3.0.4 (continued)

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 entry into 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, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.~~

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."

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool (the north pool, south pool, and canal pool) meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is given in the USAR, Section 9.5 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the USAR, Section 9.3 (Ref. 2). The assumptions of the fuel handling accident are given in the USAR, Section 14.2.1 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant 2 hour TEDE dose per person at the exclusion area boundary is $<$ 25% of the 10 CFR 50.67 (Ref. 5) limits.

According to Reference 3, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be $<$ 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. Based on studies performed to confirm the stripping efficiency of the spent fuel pool water with laboratory tests (Ref. 6), the use of the $\geq$ 23 ft decontamination factor is acceptable.

The spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel pool water level is required to be $\geq$ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for irradiated fuel movement within the spent fuel pool.

BASES

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.

ACTIONS

A.1

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

~~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 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

REFERENCES

1. USAR, Section 9.5.
2. USAR, Section 9.3.
3. USAR, Section 14.2.1.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.
6. Malinowski, D.D., Bell, M.J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The spent fuel pool at Kewaunee Power Station (KPS) is comprised of three separate pools, a large south pool, a smaller north pool, and a third pool designated as the canal pool and a fuel transfer canal that are connected to one another to allow for movement of spent fuel (Ref. 1). The original spent fuel pool storage racks in the north and south pools have been replaced with high-density spent fuel racks, permitting a larger number of spent fuel assemblies to be stored in the pool. An additional storage pool (canal pool) was created at the north end of the fuel transfer canal. The spent fuel in the canal pool is limited to assemblies that have been discharged from the reactor core prior to or during the 1984 refueling outage. The north and south pools (combined), with 990 storage positions, are designed to accommodate new fuel with a maximum enrichment of 4.9976 wt% U-235, or spent fuel regardless of the discharge fuel burnup. The canal pool, with 215 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the Acceptable Domain according to Figure 3.7.15-1, in the LCO. Fuel assemblies within the Unacceptable Domain region of Figure 3.7.15-1 shall be stored in the north and south pools (combined).

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel pool is based on the use of unborated water, which maintains each separate pool and the transfer canal in a subcritical condition during normal operation with the three separate pools fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is the inadvertent placement of fresh (unirradiated) fuel assembly into a location restricted to a burned assembly. This could potentially increase the reactivity of the north and south combined pools. To mitigate this postulated criticality related accident, boron is dissolved in the pool water. Safe operation of all three separate pools with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

BASES

APPLICABLE SAFETY ANALYSES	Most accident conditions do not result in an increase in the activity of either of the two areas (the north and south pool racks and the canal pool racks). Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in both areas. The postulated accidents are basically of two types. A fuel assembly could be incorrectly loaded in the canal pool racks (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded storage rack. This could have a small positive reactivity effect. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses are provided in Holtec Report No. HI-992208 (Ref. 3). This report concluded a minimum of 240 ppm of boron is sufficient to ensure criticality does not occur during the worst case fuel loading accident in the spent fuel pool racks (i.e., a fuel assembly misloaded in the canal pool racks). This report also concluded that no boron was necessary to ensure subcriticality during a dropped fuel assembly event in the spent fuel pool. The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The spent fuel pool boron concentration is required to be ≥ 240 ppm. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

BASES

ACTIONS

A.1, A.2.1, and A.2.2

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

~~If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. USAR, Section 9.5.2.3.
2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
3. License Amendment Request 167, dated 11/18/99, Attachment 5, Holtec Report No. HI-992208, "Licensing Report for Storage Capacity Expansion of the Kewaunee Nuclear Power Plant."

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel pool at Kewaunee Power Station (KPS) is comprised of three separate pools, a large south pool, a smaller north pool, and a third pool designated as the canal pool and a fuel transfer canal that are connected to one another to allow for movement of spent fuel (Ref. 1). The original spent fuel pool storage racks in the north and south pools have been replaced with high-density spent fuel racks, permitting a larger number of spent fuel assemblies to be stored in the pool. An additional storage pool (canal pool) was created at the north end of the fuel transfer canal. The spent fuel in the canal pool is limited to assemblies that have been discharged from the reactor core prior to or during the 1984 refueling outage. The north and south pools (combined), with 990 storage positions, are designed to accommodate new fuel with a maximum enrichment of 4.9776 wt% U-235, or spent fuel regardless of the discharge fuel burnup. The canal pool, with 215 storage positions, is designed to accommodate fuel of various initial enrichments which have been discharged prior to or during the 1984 refueling outage and which have accumulated minimum burnups within the Acceptable Domain according to Figure 3.7.15-1, in the accompanying LCO. ~~New fuel assemblies,~~ ~~s~~Spent fuel assemblies which have been discharged after the 1984 outage, or spent fuel assemblies within the Unacceptable Domain region of Figure 3.7.15-1 shall be stored in the north and south pools (combined).

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel pool is based on the use of unborated water, which maintains each separate pool and the transfer canal in a subcritical condition during normal operation with the three separate pools fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario ~~is~~ was the inadvertent placement of a fresh (unirradiated) fuel assembly into a location restricted to a burned assembly. This could have potentially increased the reactivity of the north and south combined pools. To mitigate this postulated criticality related accident, boron is dissolved in the pool water. Safe operation of all three separate pools with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO.

Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

BASES

APPLICABLE SAFETY ANALYSES	The hypothetical accidents can only take place during or as a result of the movement of an assembly (Ref. 3). By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO. The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.15-1, in the accompanying LCO, ensures the k_{eff} of the spent fuel pool will always remain < 0.95, assuming the pool to be flooded with unborated water. Irradiated fuel assemblies discharged prior to or during the 1984 refueling outage with a combination of burnup and initial nominal enrichment in the Acceptable Domain of Figure 3.7.15-1 are allowed to be stored in the transfer canal spent fuel pool or the north and south combined spent fuel pools. ~~New fuel assemblies, i~~Irradiated fuel assemblies discharged after the 1984 refueling outage, or spent fuel assemblies not in the Acceptable Domain of Figure 3.7.15-1 shall be stored in the north and south pools (combined). New fuel is no longer stored onsite.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pool.
ACTIONS	A.1 ~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~ When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with LCO 3.7.15 and Figure 3.7.15-1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with LCO 3.7.15 and Figure 3.7.15-1. ~~If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.15.1 This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 in the accompanying LCO.
REFERENCES	1. USAR, Section 9.5.2.3. 2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A). 3. USAR, Section 14.2.1.