ATTACHMENT 1

LICENSE AMENDMENT REQUEST 244: PROPOSED REVISION TO RADIOLOGICAL ACCIDENT ANALYSIS AND CONTROL ROOM ENVELOPE HABITABILITY TECHNICAL SPECIFICATIONS

DISCUSSION OF CHANGE, SAFETY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION, AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

PROPOSED REVISION TO RADIOLOGICAL ACCIDENT ANALYSIS AND CONTROL ROOM ENVELOPE HABITABILITY TECHNICAL SPECIFICATIONS

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PROPOSED REVISION TO RADIOLOGICAL ACCIDENT ANALYSIS AND CONTROL ROOM ENVELOPE HABITABILITY TECHNICAL SPECIFICATIONS

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to the Kewaunee Facility Operating License Number DPR-43. The proposed amendment would modify the Kewaunee Power Station (KPS) Operating License, Technical Specifications (TS), and current licensing basis to incorporate changes to the current radiological accident analysis (RAA) of record. The proposed amendment would also implement a commitment (reference 3) made in conjunction with the KPS response to NRC Generic Letter (GL) 2003-01, “Control Room Habitability,” to submit proposed changes to the KPS TS based on the final approved version of TSTF-448 (reference 5) regarding control room habitability. The proposed changes would also incorporate the following Technical Specification Task Force travelers (TSTF’s): TSTF-51, TSTF-490, and TSTF-312.

1.1 Implementation of New Radiological Accident Analysis

The first set of proposed changes would revise the KPS Operating License (OL) by modifying the Technical Specifications (TS), and the current licensing basis. These changes would incorporate a revision to the current radiological accident analysis (RAA) of record, which is provided in Attachment 4 of this submittal. DEK requests NRC review and approval of the revised RAA in accordance with 10 CFR 50.59(c)(2) because incorporation of the revised RAA involves more than a minimal increase in the consequences of an accident previously evaluated. Tables 3.2-1 and 3.2-2 of this attachment provide the current and proposed new design basis accident (DBA) calculated radiological consequences.

The details concerning the methods, assumptions, and results of the proposed new RAA are provided in Attachment 4. The proposed new RAA affects the following eight DBAs described in the USAR:

- Main Steam Line Break (MSLB) Accident
- Locked Rotor Accident (LRA)
- Rod Ejection Accident (REA)
- Steam Generator Tube Rupture (SGTR)
- Loss-of-Coolant Accident (LOCA)
- Waste Gas Decay Tank (WGDT) Rupture
- Volume Control Tank (VCT) Rupture
- Fuel Handling Accident (FHA)

In addition, consistent with the revised RAA, DEK is proposing Technical Specification changes which adopt TSTF-312, Revision 1, “Administratively Control Containment Penetrations” (reference 7) and modify the TS definition of Dose Equivalent Iodine 131 consistent with TSTF-490, Revision 1 (reference 8).

1.2 Implementation of GL 2003-01, “Control Room Habitability” Commitment

The second set of proposed changes would add a new License Condition to the KPS Operating License (OL), add a new CREH program, and modify the TS to incorporate changes related to Control Room Envelope Habitability (CREH), consistent with adoption of TSTF-448.

NRC GL 2003-01, “Control Room Habitability,” (reference 2) informed licensees that existing TS surveillance requirements (SRs) for systems necessary to maintain CREH might not be adequate at some facilities. Specifically, the Generic Letter noted that tracer gas test results at some facilities had indicated that a differential pressure (ΔP) test is not a reliable surveillance method for demonstrating control room envelope (CRE) integrity.

The Technical Specification Task Force and the Nuclear Energy Institute Control Room Habitability Task Force developed proposed changes to the Improved Standard Technical Specifications (ISTS) (NUREGs-1430 through 1434) to address the CREH issue by:

1. Replacing the ΔP surveillance with a tracer gas test surveillance;
2. Adding a TS Action to address situations when the CRE is inoperable, and;
3. Instituting a CREH program to ensure that CREH is maintained.

The proposed changes would revise the KPS OL and TS to adopt the following TSTF’s:

1. TSTF-448, Revision 3, “Control Room Habitability” (Reference 5).
2. TSTF-51, Revision 2, “Revise Containment Requirements during Handling of Irradiated Fuel and Core Operations” (Reference 10).

The proposed changes will ensure CRE habitability is maintained by establishing plant-specific CREH TS’s and a new CREH program. The design of the KPS control room results in some differences between the approved wording in these TSTFs and the wording proposed in this LAR. For example, the KPS control room is a neutral-pressure control room and is not intentionally pressurized during accident conditions.

This LAR proposes to adopt the NRC CLIIP for TSTF-448 pursuant to the requirements of 10 CFR 50.90. Adoption of TSTF-448 fulfills a commitment to submit proposed changes to the TS based upon the final approved version of TSTF-448. This commitment was made in the KPS response to NRC GL 2003-01(reference 5).

1.3 General Information

Associated TS Bases changes will be made in conjunction with the TS changes proposed in this amendment request. The TS Bases changes will be implemented at the same time as the proposed TS changes. The TS Bases changes are provided in Attachment 3 for information.

Several NRC commitments associated with the changes described in Section 2.0 have been made in this amendment request. These commitments are described in detail in Section 4.0 of this Attachment and in the cover letter to this submittal.

2.0 PROPOSED CHANGES

The proposed changes in this amendment request are separated into two sections. Section 2.1 describes the proposed changes related to implementation of the revised RAA. Section 2.2 describes the proposed changes related to adoption of TSTF-448. Marked-up copies of the current KPS TS pages are provided in Attachment 2. Marked-up copies of the current TS Bases pages are provided in Attachment 3 for information. An evaluation of each of the proposed changes described below is provided in Section 4 of this Attachment.

2.1 Proposed Changes to Incorporate Revised Radiological Accident Analysis

The proposed amendment would modify current KPS TS requirements as follows:

1. TS 3.4.16, "RCS Specific Activity," and TS 3.7.16, "Secondary Specific Activity," would be revised to incorporate new specific activity limits, consistent with the revised RAA.
2. TSTF-51 would be adopted to replace the term "irradiated fuel" with the term "recently irradiated fuel" in several KPS TS.
3. TS 3.9.6, "Containment Penetrations," regarding Containment Closure during Refueling Operations, would be revised for two purposes:
    a. To allow the containment equipment hatch be open during handling of recently irradiated fuel provided it is capable of being closed.
    b. To permit containment penetration air paths to be un-isolated under administrative control while handling recently irradiated fuel, consistent with TSTF-312 and TSTF-51.
4. The definition of Dose Equivalent Iodine would be revised, consistent with TSTF-490 and the revised RAA in Attachment 4.

A detailed description of each of the above proposed changes is provided below.

2.1.1. Revise Specific Activity Limits

Changes to TS 3.4.16, "RCS Specific Activity"

This amendment proposes to revise TS 3.4.16, "RCS Specific Activity," to incorporate new RCS specific activity limits. The proposed new limits are consistent with the revised RAA in Attachment 4. The proposed changes are as follows:

1. TS 3.4.16 Required Action A.1 and Condition C each specify a DOSE EQUIVALENT Iodine-131 (DEI) specific activity limit of ≤ 20 μCi/gm. DEK is proposing to reduce the reactor coolant DEI specific activity limit in TS 3.4.16 Required Action A.1 and Condition C from the current limit of ≤ 20 μCi/gm to a new limit of ≤ 10 μCi/gm.

2. SR 3.4.16.1 requires verification that DOSE EQUIVALENT Xenon-133 (DEX) specific activity is ≤ 595 μCi/gm on a 7-day frequency. DEK is proposing to reduce the reactor coolant DEX specific activity limit in SR 3.4.16.1 from ≤ 595 μCi/gm to ≤ 16.4 μCi/gm.
3. SR 3.4.16.2 requires verification that DOSE EQUIVALENT Iodine-131 (DEI) specific activity is ≤ 1.0 μCi/gm on a 14-day frequency, and between 2 and 6 hours after a thermal power change of ≥ 15% of rated thermal power within a one hour period. DEK is proposing to reduce the reactor coolant DEI specific activity limit in SR 3.4.16.2 from ≤ 1.0 μCi/gm to ≤ 0.1 μCi/gm.

Changes to TS 3.7.16, "Secondary Specific Activity"

LCO 3.7.16 currently specifies secondary coolant specific activity shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT Iodine-131 (DEI). In addition, SR 3.7.16.1 requires verification that secondary coolant specific activity is ≤ 0.10 μCi/gm DEI on a 31-day frequency. DEK is proposing to reduce the secondary coolant DEI specific activity limit in LCO 3.7.16 and SR 3.7.16.1 from ≤ 0.10 μCi/gm to ≤ 0.05 μCi/gm.

A markup of the affected TS pages is provided in Attachment 2.

2.1.2 Adoption of TSTF-51

DEK is proposing to adopt Technical Specification Task Force Traveler (TSTF)-51, "Revise containment requirements during handling of irradiated fuel and core alterations" (reference 10). DEK proposes to change the wording, "During [Suspend] movement of irradiated fuel assemblies," to, "During [Suspend] movement of recently irradiated fuel assemblies," in the TS listed in Table 2-1 below. A markup of the affected TS pages is provided in Attachment 2.

TABLE 2-1 List of Technical Specification Sections Affected by Adoption of TSTF-51	
Technical Specification	**Sections Affected by Adoption of TSTF-51**
3.3.6 Containment Purge and Vent Isolation Instrumentation	− Note applicable to Condition C − Table 3.3.6-1, Footnote (a)
3.3.7 Control Room Post-Accident Recirculation (CRPAR) System Actuation Instrumentation	− Condition D and Required Action D.1 − Table 3.3.7-1, Footnote (a)
3.7.10 Control Room Post-Accident Recirculation (CRPAR) System	− Proposed new LCO Note − Applicability Statement − Condition D and Required Action D.2 − Condition E and Required Action E.1
3.7.11 Control Room Air Conditioning (CRAC) Alternate Cooling System	− Applicability Statement − Condition C and Required Action C.2 − Condition D and Required Action D.1
3.8.2 AC Sources – Shutdown	− Applicability Statement − Required Action A.2.1 − Required Action B.1
3.8.5 DC Sources – Shutdown	− Applicability Statement − Required Action A.1
3.8.8 Inverters – Shutdown	− Applicability Statement − Required Action A.1
3.8.10 Distribution Systems – Shutdown	− Applicability Statement − Required Action A.2.1
3.9.6 Containment Penetrations	− Applicability Statement − Required Action A.1

2.1.3 Revise TS 3.9.6, Containment Penetrations

DEK is proposing changes to TS 3.9.6, "Containment Penetrations," which will provide the flexibility to open containment penetration flow paths under administrative controls during refueling operations. The proposed changes to TS 3.9.6 are summarized below:

1. Incorporate the term "recently" into the phrase "During [Suspend] movement of [recently] irradiated fuel" in the TS 3.9.6 Applicability statement and Required Action A.1, consistent with TSTF-51 (see Section 2.1.2 above).
2. Change LCO 3.9.6.a to allow the containment equipment hatch to be open during handling of recently irradiated fuel when measures are in place which ensure the capability to close equipment hatch in the event of a fuel handling accident.
3. Incorporate a new Note, applicable to LCO 3.9.6.c, which would allow penetration flow paths providing direct access from the containment to outside atmosphere to be opened under administrative controls, consistent with adoption of TSTF-312 (reference 7).

A markup of the affected TS pages is provided in Attachment 2.

2.1.4 Revise TS 1.1 Definition of Dose Equivalent I-131

DEK proposes to change the TS 1.1 definition of Dose Equivalent I-131. DEK proposes to revise the definition of Dose Equivalent I-131 to reference Table 2.1 of FGR No. 11 as the source of thyroid committed dose equivalent (CDE) dose conversion factors based on the use of this table in the revised RAA in Attachment 4. A mark-up of the TS 1.1 definition of Dose Equivalent I-131 is provided in Attachment 2. This proposed change is consistent with TSTF-490, Revision 1 (reference 8).

2.2 Proposed Changes to Establish Control Room Envelope Habitability Requirements

The proposed amendment would modify the current Operating License (OL) and affect TS requirements related to Control Room Envelope Habitability (CREH) as discussed below. An evaluation of each of the proposed changes described below is provided in Section 4.2 of this Attachment.

2.2.1. Add Control Room Envelope Habitability Program

DEK is proposing to add a new section to TS 5.5, "Programs and Manuals." The new section, TS 5.5.17, "Control Room Envelope Habitability Program," would establish requirements for a CRE habitability program consistent with adoption of TSTF-448 (reference 5). The wording of the proposed TS 5.5.17 is shown in Attachment 2.

2.2.2. Modify TS 3.7.10, Control Room Post-Accident Recirculation (CRPAR) System

DEK is proposing to modify TS 3.7.10 consistent with adoption of TSTF-448. The proposed changes to TS 3.7.10 are described below and shown in Attachment 2.

1. The existing NOTE in LCO 3.7.10 would be modified to change the current wording from; "*The control room boundary may be opened intermittently under administrative control*" to; "*The control room envelope (CRE) boundary may be opened intermittently under administrative control*."
2. A new NOTE would be added to LCO 3.7.10 which states; "The CRE shall be isolated during movement of recently irradiated fuel assemblies."
3. The current APPLICABILITY for LCO 3.7.10 is Modes 1-6, and during movement of irradiated fuel assemblies. The APPLICABILITY would be changed to Modes 1-4, and during movement of recently irradiated fuel assemblies. Consistent with this change, TS 3.7.10, Condition D and Condition E are also modified by removing Mode 5 and 6 applicability.
4. TS 3.7.10, Condition A wording would be modified from "*One CRPAR Train inoperable*" to "*One CRPAR Train inoperable for reasons other than Condition B*."
5. TS 3.7.10, Condition B currently provides a Required Action when two CRPAR trains are inoperable due to an inoperable CRE boundary in Modes 1-4. The current Required Action B.1 is to restore the CRE boundary to operable status within 24 hours. Condition B and its associated Required Action B.1 and Completion Time would be replaced with a new Condition B. The new Condition B would provide required actions and completion times when one or more CRPAR trains are inoperable due to an inoperable CRE boundary in Modes 1-4.

   The new Condition B would include three new Required Actions when one or more CRPAR trains are inoperable due to an inoperable CRE boundary in Modes 1-4. The three new Required Actions (B.1, B.2, and B.3) would require; (B.1) immediate initiation of action to implement mitigating actions; (B.2) verification that mitigating actions ensure CRE occupant exposures to radiological, chemical and smoke hazards will not exceed limits within 24 hours and; (B.3) restoration of the CRE boundary to operable status within 90 days.
6. TS 3.7.10, Condition E currently requires immediate suspension of movement of irradiated fuel assemblies when two CRPAR trains are inoperable in Modes 5 and 6 and during movement of irradiated fuel assemblies. The current Condition E would be modified by deleting Mode 5 and 6 applicability (see item 3 above) and adding the word "recently" so that the resulting Condition would read; "*Two CRPAR trains inoperable during movement of recently irradiated fuel assemblies*." In addition, Condition E would be expanded to include situations where the Required Actions and associated Completion Times of Condition B are not met during movement of recently irradiated fuel assemblies. The Required Action of Condition E would be

changed from *"suspend movement of irradiated fuel assemblies,"* to *"suspend movement of recently irradiated fuel assemblies."* The Completion Time of Condition E would remain unchanged.

7. A new Surveillance Requirement 3.7.10.4 would be added. New SR 3.7.10.4 would require unfiltered air inleakage testing of the Control Room Envelope in accordance with the proposed new Control Room Envelope Habitability Program discussed in Section 2.2.1 above.

2.2.3. Delete TS Requirements for Control Room Vent Radiation Monitor

TS 3.3.7, "Control Room Post Accident Recirculation (CRPAR System Actuation Instrumentation," contains requirements associated with actuation instrumentation for the Control Room Post Accident Recirculation (CRPAR) system. One of the instruments included in TS 3.3.7 is the control room vent radiation monitor (radiation monitor R-23). This instrument is listed in Table 3.3.7.1, as Function 2, "Control Room Vent Radiation Monitor." This Function would be deleted from the TS (and relocated to the KPS Technical Requirements Manual) because radiation monitor R-23 is not credited in the revised RAA.

Consistent with deletion of Table 3.3.7.1, Function 2, the portion of Condition B which states "*<u>OR</u> Control Room Vent Radiation Monitor inoperable*," would be deleted and SR 3.3.7.1, SR 3.3.7.2, and 3.3.7.4 would be deleted because these SRs are only applicable to Function 2. DEK will relocate these requirements to the KPS Technical Requirements Manual.

In addition, Table 3.3.7-1 would be modified to delete Mode 5 and 6 applicability for the CRPAR System Actuation Instrumentation, consistent with the proposed changes to TS 3.7.10 discussed in Section 2.2.2 (item 3).

2.2.4. License Condition for Implementation of TSTF-448 Requirements

The proposed amendment would add a new license condition to the KPS Operating License consistent with adoption of TSTF-448. The proposed new license condition would establish schedule requirements for initial Control Room Envelope (CRE) testing and assessment. The new license condition is shown in Attachment 2.

2.3 Summary of Proposed Changes

This amendment would revise the TS to adopt TSTF-448 and to fulfill a commitment provided in DEK's response to GL 2003-01.

Adoption of other TSTFs is also proposed, including TSTF-51 and TSTF-312. The associated changes to the TS Bases are included in Attachment 3 for information.

DEK is also proposing changes to the TS and USAR to incorporate a revision to the current RAA of record. The changes would affect eight design basis accident analyses described in the USAR. The details concerning the assumptions, methods, and results of the proposed new RAA are provided in Attachment 4.

3.0 BACKGROUND

3.1 Plant / System Description

Kewaunee Power Station (KPS) is a 2-loop Westinghouse pressurized water reactor design nuclear electrical generating station. The reactor coolant system (RCS) consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a circulating pump (reactor coolant pump, RCP), loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the RCS is shown in the Updated Safety Analysis Report (USAR) Figure 4.2-1.

The Containment System consists of two separate structures: the Reactor Containment Vessel and the Shield Building. The Reactor Containment Vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor pressure vessel, the steam generators, reactor coolant pumps, the reactor coolant loops, the accumulators of the Safety Injection System, the reactor coolant pressurizer, the pressurizer relief tank and other branch connections of the Reactor Coolant System.

The Reactor Containment Vessel is completely enclosed by the Shield Building. The Shield Building has the shape of a right circular cylinder with a shallow dome roof. A 5-foot annular space is provided between the Reactor Containment Vessel and the Shield Building. Clearance at the roof of the Shield Building is 7 feet. The Reactor Containment Vessel, including penetrations, is designed for low leakage. The total containment consists of two systems as shown in USAR Figure 5.1-1. The Reactor Containment Vessel is also referred to as the Primary Containment System. It is designed to confine the radioactive materials that could be released by accidental loss of integrity of the Reactor Coolant System pressure boundary. Systems directly associated with the Primary Containment System include the Internal Containment Spray, Containment Air Cooling and Containment Isolation Systems.

The Secondary Containment System consists of two structures and their associated ESF Systems: 1) the Shield Building and its associated ESF System, the Shield Building Ventilation (SBV) System; and 2) the Auxiliary Building Special Ventilation Zone and its associated ESF System, the Auxiliary Building Special Ventilation (ASV) System. The entire envelope that comprises the Shield Building boundary has been constructed to minimize air leakage across the boundary.

Steam from each of the two steam generators supplies the turbine, where the steam expands through the double-flow high-pressure turbine, and then flows through moisture separator reheaters to two double-flow low-pressure turbines in tandem. The Main Steam System directs steam in a 30-inch pipe from each of the two steam generators within the reactor containment through a swing-disc type isolation valve (main steam isolation valve, MSIV) and a swing-disc type non-return valve to the turbine stop and control valves. The isolation and non-return valves are located outside of the containment. The two steam lines are interconnected near the turbine.

The main steam isolation valves utilize a swing-disc, which is normally held out of the main steam flow path by an air piston. These valves are closed by steam flow (aided by a spring) upon receipt of a signal from the steam line isolation protection system. These isolation valves are designed to close within five seconds after a trip signal is received. The non-return valves prevent reverse flow of steam. If a steam line ruptures between a non-return valve and a steam generator, the affected steam generator will blow down. The non-return valve in the steam line will eliminate blowdown (reverse flow) from the other steam generator.

If the condenser heat sink is not available or steam dump rate exceeds the steam dump system capacity, excess steam generated as a result of RCS sensible heat and core decay heat is discharged to the atmosphere. There are five 6-inch by 10-inch code safety valves located on each of the two 30-inch main steam lines outside the reactor containment and upstream of the isolation and non-return valves. Discharge from these safety valves is to atmosphere through vent lines. In addition, one power operated relief valve (PORV) is provided in each main steam line, which is capable of releasing the sensible and core decay heat to the atmosphere. These valves are automatically controlled by pressure or may be manually operated from the main control board. The PORV's may also be used to release the steam generated during reactor physics testing and plant hot standby operation, if the condenser is not available.

Excess steam generated by the RCS can be bypassed to the condenser by means of two 18-inch main steam dump lines (one for each condenser) that feed three 8-inch lines to each condenser. In addition, three atmospheric dump valves are provided on each main steam line in a common header downstream of the non-return valves.

3.1.1 Control Room Air Conditioning (Ventilation) System

The KPS control room contains the controls and instrumentation necessary for safe operation of the plant under normal and accident conditions.

Sufficient design features (shielding, distances, containment integrity and filtration systems) are provided to assure that control room personnel are not subjected to doses, under postulated accident conditions during occupancy of the control room, which would exceed 5 rem total effective dose equivalent (TEDE) for 30 days following the accident.

The control room air conditioning (CRAC) system is designed to provide a reliable means of cooling and filtering air supplied to the control and relay rooms under both normal and post-accident conditions. The CRAC system is shown in the KPS USAR Figure 9.6-6.

The CRAC system is normally in operation providing cooled and filtered air to the control room and relay room. There is normally a 15 percent fresh air makeup to the control room from the auxiliary building air conditioning unit air intake. Although the normal fresh air intake is from the auxiliary building air conditioning intake, an alternate source of fresh air is provided from the auxiliary building ventilation system intake should conditions warrant. The makeup air passes through roughing filters, cooling coils, and fans into one of the two 100 percent capacity control room air conditioning units and is then distributed to the control and relay rooms. Heating coils supplied from the auxiliary building hot water converter provide for comfort heating. Service water can be aligned directly to the cooling coils in the air handler in the event that both chilled water units are not available.

The CRAC system provides a large percentage of recirculated air while in the normal mode of operation. Process radiation monitor R-23 continuously monitors CRAC system recirculation air for an indication of airborne activity entering through the ventilation system. The R-23 detector is a beta-sensitive plastic scintillator that is mounted in the air supply duct after the air handling unit. Radiation monitor R-23 is a single train, non-safety related circuit. Readout is in the control room on multipoint recorders and at a rate meter station with a high-low alarm setting.

If a high radiation condition exists, the R-23 circuit initiates closure of the outside air intake dampers and starts a CRAC subsystem called the control room post accident recirculation (CRPAR) system. In addition, the control room is provided with an area radiation monitor channel R-1. Radiation monitor R-1 monitors the control room area for radiation and alarms in the control room, alerting the operators to the abnormal condition. Neither R-1 nor R-23 is credited in the proposed RAA (see Attachment 4 for details).

The CRPAR system consists of two trains. Each train consists of a CRPAR fan and a filter unit. Each filter unit consists of a pre-filter, HEPA filter, and a charcoal filter. Starting a CRPAR fan initiates the following:

- Closes recirculation dampers ACC-2 and ACC-5.
- Opens recirculation dampers ACC-3A (A train), ACC-3B (B train).
- Starts the corresponding train CRAC fan.

A CRPAR train starts upon receipt of a corresponding safety injection signal, steam exclusion signal, or by manual initiation of the CRPAR fan. Radiation monitor R-23, as a single channel, initiates both trains of the CRPAR system.

A safety injection signal is generated by any of the following:

- Low pressurizer pressure (sensed on 2 out of 3 channels); this signal can be manually blocked when pressurizer pressure (sensed on 2 out of 3 channels) is below a preset value.
- High reactor containment vessel pressure (sensed on 2 out of 3 channels).
- Low steam line pressure per loop (sensed on 2 out of 3 channels); this signal can be manually blocked when pressurizer pressure (sensed on 2 out of 3 channels) is below a given set point.

Two trains of SI instrumentation are provided; A and B train. A signal from either SI train causes an isolation of the KPS control room and initiates the associated CRPAR fan and filtration unit train.

In the event of a postulated High Energy Line Break (HELB), steam entry into steam exclusion zones is blocked through ventilation ductwork pathways by design. Approximately 500 feet of ventilation system ductwork has been reinforced to prevent collapse in the event of a HELB, which might allow steam intrusion from one zone to another. Thirty-eight steam exclusion dampers are provided to block ventilation system ducts. Other penetrations between steam environments and steam exclusion zones are also blocked. In these cases, damper actuation is accomplished by use of a proven safeguarding system of thirty (30) temperature sensing elements (Resistance Temperature Detectors (RTDs)) at ten locations using 2/3 logic, with a setpoint of 140°F. A safeguarding rack provides proper electrical and physical separation. This system will function any time high temperature conditions are present from postulated breaks in any of the high-energy line systems.

3.1.2 Containment Equipment Hatch and Personnel Airlocks

The containment equipment hatch and personnel air locks were fabricated from welded steel and furnished with double-gasketed flanges and bolted dished doors. Provision is made to pressure-test the space between the double gaskets. The equipment hatch is shielded by a 2-foot 6-inch thick concrete shadow shield.

The equipment hatch (i.e. the containment building inner equipment hatch) is opened and closed by means of a trolley and jactuator system. There are no electrical or compressed air requirements for movement of the equipment hatch. Chain drives or pulls are provided that can be operated by a single individual to open and close the equipment hatch. When in the closed position, the equipment hatch is held in place by 12 bolts.

To close the equipment hatch, the operator simply moves the hatch into the correct position for closure using the chain pulls on the jactuator (to move the hatch into or out of the containment wall) and the trolley (to move the hatch into alignment with the containment opening using side to side motion). Then 12 nuts and washers are placed into the indicated positions on the equipment hatch and tightened in place. For containment closure requiring a leak tight connection, four nuts are torqued to the proper value.

Two containment personnel air locks are provided. Each personnel air lock is a double-door welded steel assembly. Quick-acting type equalizing valves are provided to equalize pressure in the air lock when personnel enter or leave the Reactor Containment Vessel. Provision is made to pressurize the air locks for periodic leak-rate tests.

The two doors in each personnel air lock are interlocked to prevent both doors from being opened simultaneously, and to ensure that one door is completely closed before the other door can be opened. When one air lock door is opened, the other door is automatically locked and cannot be opened until the open door is closed. Remote indicating lights in the control room indicate the door operational status. Provision is also made to permit by-passing the door interlocking system with a special tool, to allow both doors to be left open during plant cold shutdown conditions. Each air lock door hinge can be adjusted to assist in proper seating. A lighting and communication system, which can be operated from an external emergency power supply, is provided within each air lock.

The equipment hatch and personnel air locks are supported entirely by the Reactor Containment Vessel and are not connected either directly or indirectly to any other structure. The materials for penetrations, including the personnel access air locks and the equipment access hatch, conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Code Case 1392, Revision 0.

3.2 Applicable Licensing Bases History

3.2.1 Incorporation of TMI Action Item Requirements

NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.4, "Control Room Habitability Requirements," required licensees to assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gas and that the plant can be safely operated or shutdown under design basis accident conditions. In response to NUREG-0737, Item III.D.3.4, the KPS staff performed a review of post-accident control room habitability and transmitted the results to the NRC (reference 21).

Wisconsin Public Service Corporation (WPSC), the licensee at the time, took four exceptions to the NRC Standard Review Plan acceptance criteria. The exceptions were:

1. Requirements for the storage of food supplies in the control room.
2. Requirement for the storage of potassium iodide tablets in the control room.
3. Requirement for redundancy of radiation monitors in the control room normal ventilation system air intake.
4. Requirement to perform a toxic gas, ammonia spill, analysis to determine the effects on control room habitability.

In the NRC's safety evaluation (SE) (reference 25) for NUREG-0737, Item III.D.3.4, the NRC stated that they reviewed the KPS submittals and evaluated them using the criteria of Standard Review Plan (NUREG-0800) (SRP) Sections 2.2.1, 2.2.2, 2.2.3, and 6.4, and RGs 1.78 and 1.95. The NRC determined that the control room habitability systems were acceptable and would provide a safe, habitable environment within the control room under design basis accident radiation and toxic gas conditions. The NRC concluded that the design meets the criteria identified in NUREG-0737 and is acceptable.

The NRC staff accepted WPSC's position for exceptions 1 and 2 that stores of food and potassium iodide need not be kept within the control room, and concluded it was sufficient that they be readily available from nearby sources. WPSC resolved the fourth exception by reporting their re-appraisal of protection of control room habitants from toxic gas releases using RG 1.78 guidance, which concluded that off-site toxic gas releases would not result in control room air concentrations above acceptance levels of RG 1.78. Exceptions 1, 2, and 4 are not addressed or evaluated further by this LAR and are considered still in effect.

The NRC accepted WPSC's position for exception 3 based on the condition that for radioactive releases, at least one other radiation monitor would alarm in the control room or a control room ventilation system isolation signal would occur, such that the single radiation monitor in the air intake (i.e., radiation monitor R-23) would not be the sole means of isolating that system (reference 25).

The changes proposed in this LAR modify the basis for NRC accepting exception 3 above. The revised RAA in Attachment 4 does not rely on radiation monitor R-23. Radiation monitor R-23 is a single train, non-safety related instrument and is located in the control room normal ventilation system air intake. Previously, the analysis performed for control room habitability relied on radiation monitor R-23 to monitor the control room ventilation intake air for radiation, initiate closure of the outside air intake, and start the CRPAR system. Using a single channel to perform the control room isolation function did not meet the redundancy guidance in NUREG-0737 for radiation monitors used in the control room normal ventilation system air intake. To compensate for this, KPS credited other area radiation monitors to alarm in the control room upon reaching the radiation monitor setpoint.

Exception 3 above will be modified by this LAR and replaced with analysis and controls that do not credit the function of radiation monitor R-23. Previous analysis (LRA and FHA) credited the operation of R-23 with a backup to manual actions based on communications and other radiation monitors that alarm in the control room. The revised RAA in Attachment 4 credits reasonable operator actions (LRA and FHA only), and a proposed new TS 3.7.10 requirement to pre-isolate the control room during movement of recently irradiated fuel assemblies (FHA only).

3.2.2 Adoption of Alternate Source Term (AST)

In 2002, the KPS design basis accident RAA's were revised to support implementation of AST. Nuclear Management Company (NMC), the licensee at that time, requested a revision to the KPS design-basis accident RAA's to support implementation of AST as described in RG 1.183. The results of the revised RAA's demonstrated that post-accident doses remained below the appropriate limits of 10 CFR 50.67 (reference 22). The revised RAA's assumed a control room unfiltered inleakage (UFI) of 200 cubic feet per minute (cfm) based on previous analysis and testing of the CRE boundary. The revised RAA's determined that sufficient design features are in place (shielding, distances, and containment integrity and filtration systems) to assure that control room personnel will not be subjected to post-accident doses that exceed established acceptance criteria.

Subsequently, the NRC found the revised RAA's acceptable and issued an amendment approving the use of the AST methodology at KPS (reference 17).

3.2.3 Application for Stretch Power Uprate

Following the amendment approving the use of the AST methodology, a new RAA was developed to support the KPS stretch power uprate. This new RAA satisfied the control room dose acceptance criteria of 5.0 rem TEDE and was approved as part of the Kewaunee stretch power uprate (KPS License Amendment 172 (reference 18)) in February of 2004.

3.2.4 Incorporation of Radiological Accident Analysis Changes to Account for Measured Control Room Inleakage

In December 2004, tracer gas testing was performed to confirm the unfiltered in-leakage into the KPS control room envelope (CRE) (reference 20). The tracer gas in-leakage test showed that the radiological accident analysis CRE unfiltered in-leakage assumption of 200 cfm was not conservative (i.e. measured in-leakage was greater than assumed in-leakage). An operability determination was performed. The operability determination specified revised interim administrative limits for containment leak rate, reactor coolant system activity, and carbon filter absorption efficiency affecting radiological source and potential radiological release pathways. These administrative limits compensated for the difference between the assumed and measured control room in-leakage and were incorporated into the appropriate plant procedures. The administrative limits, developed from radiological accident analysis sensitivity cases, ensured that the radiological dose consequences remained within the licensing basis acceptance criteria of 10 CFR 50.67, including the acceptance criteria limitations of RG 1.183. No credit was taken in the operability determination for the use of self-contained breathing apparatus or potassium iodide.

The amount of air in-leakage into the CRE was evaluated using the concentration decay method under isolated conditions. Two concentration decay tests were performed to determine total unfiltered control room envelope in-leakage, one with CRPAR Train A operating, and one with CRPAR Train B operating. The tests were based on ASTM E741 requirements and were conducted to comply with NRC GL 2003-01. The ASTM E741 baseline testing results were provided to the NRC in reference 3, enclosure 1. The results obtained for total unfiltered in-leakage to the three rooms contained within the CRE are provided in Table 3-1 below.

TABLE 3-1 Control Room Inleakage Test Results		
Date of Test	**Train Tested**	**Measured Inleakage**
December 14, 2004	CRPAR Train A	409 ± 29 cfm
December 15, 2004	CRPAR Train B	447± 51 cfm

Attachment 4 provides information on how these unfiltered inleakage values are used in the revised RAA.

Because the measured CRE unfiltered in-leakage was higher than the assumed CRE unfiltered in-leakage (200 cfm) in the RAA of record at that time, the radiological accidents were re-analyzed to account for the increased CRE unfiltered in-leakage. The CRE unfiltered in-leakage assumption in the new RAA was increased to a value that bounded the measured CRE unfiltered in-leakage, including uncertainties, and also provided sufficient operating margin. An analysis of the radiological accidents using the higher CRE unfiltered in-leakage was submitted to the NRC on January 30, 2006 (reference 23) as LAR-211.

LAR-211 proposed modifications to the previously approved RAA and associated TS. This LAR proposed changes necessary to account for the difference between the CRE unfiltered in-leakage assumed in the previous RAA (200 cfm) and the CRE unfiltered in-leakage measured during tracer gas testing. The revised RAA assumed CRE inleakage to be at least 800 cfm for events that model control room isolation on a safety injection (SI) signal (e.g., Large Break Loss of Coolant, Rod Ejection, Main Steam Line Break, and Steam Generator Tube Rupture accidents) and at least 1500 cfm for events that model control room isolation on a control room radiation monitor (R-23) actuation (e.g., Fuel Handling and Locked Rotor accidents). This new RAA was approved by the NRC in KPS License Amendment 190, dated March 8, 2007 (reference 24). This is the current radiological analysis of record for KPS. The analysis results are presented in the current USAR Chapter 14.

3.3 Revised Radiological Accident Analysis Background

In accordance with a commitment made to the NRC in response to GL 2003-01 (reference 3), on September 14, 2007, DEK submitted LAR-210 to the NRC (reference 4). LAR-210 proposed incorporation of TSTF-448 into the KPS TS.

During development of a response to an NRC request for additional information in November of 2008, DEK discovered that certain input information needed to support the KPS control room atmospheric dispersion factor (X/Q value) could not be verified. Specifically, during evaluation of the CRE, it was determined that the atmospheric dispersion factor (X/Q) value of 2.93E-3 sec/m^3 used in the KPS USAR Table D.8-5 could not be verified. This led to the conclusion that the control room X/Q value used in previously approved RAAs might not have been bounding. Based on this discovery, the issue was entered into the plant corrective action program and LAR-210 was withdrawn.

Since the withdrawal of LAR-210, DEK has performed a complete reanalysis of the X/Q values associated with the RAA's. A brief synopsis of each of the proposed new KPS RAA's is provided in Attachment 4. In addition, Attachment 4 provides a comparison between the assumptions used in the currently approved RAA of record and the assumptions used in the proposed revision to the existing RAA. The table below provides the location of this information in Attachment 4 for each analyzed accident. The currently approved calculated doses and associated acceptance criteria for each analyzed accident are provided in Table 3.2-1 below. The proposed new calculated doses and associated acceptance criteria for each analyzed accident are provided in Table 3.2-2 below.

Accident	Attachment 4 Location
Large-Break Loss of Coolant Accident (LBLOCA)	Section 3.2
Fuel Handling Accident (FHA)	Section 3.3
Steam Generator Tube Rupture (SGTR)	Section 3.4
Main Steam Line Break (MSLB)	Section 3.5
Locked Rotor Accident (LRA)	Section 3.6
RCCA Ejection Accident (REA)	Section 3.7
Waste Gas Decay Tank (WGDT) Rupture	Section 3.8
Volume Control Tank (VCT) Rupture	Section 3.9

TABLE 3-2

Currently Approved Design-Basis Accident Calculated Radiological Consequences
rem TEDE (unless noted)

Design-Basis Accident	EAB	LPZ	Control Room
MSLB, Pre-existing iodine spike Dose acceptance criteria	0.030 25	0.01 25	0.70 5
MSLB, Accident-initiated iodine spike Dose acceptance criteria	0.06 2.5	0.02 2.5	2.60 5
Locked Rotor Accident Dose acceptance criteria	0.40 2.5	0.06 2.5	3.90 5
Control Rod Ejection Accident Dose acceptance criteria	0.40 6.3	0.09 6.3	4.54 5
SGTR, Pre-existing spiking Dose acceptance criteria	0.50 25	0.10 25	1.90 5
SGTR, Accident-initiated spiking Dose acceptance criteria	0.80 2.5	0.20 2.5	2.80 5
LBLOCA, total Dose acceptance criteria	0.52 25	0.09 25	4.95 5
FHA Dose acceptance criteria	0.90 6.3	0.15 6.3	4.0 5
WGDT Rupture Dose acceptance criteria	0.10 WB 0.5 WB	0.02 WB 0.5 WB	0.80 5
VCT Rupture Dose acceptance criteria	0.10 WB 0.5 WB	0.01 WB 0.5 WB	0.40 5

TABLE 3-3

Proposed New Design-Basis Accident Calculated Radiological Consequences
rem TEDE (unless noted)

Design-Basis Accident	EAB	LPZ	Control Room
MSLB, Pre-existing iodine spike Dose acceptance criteria	0.1 25	0.1 25	4.7 5
MSLB, Accident-initiated iodine spike Dose acceptance criteria	0.1 2.5	0.1 2.5	4.2 5
Locked Rotor Accident Dose acceptance criteria	0.3 2.5	0.2 2.5	4.7 5
Control Rod Ejection Accident Containment Release Pathway Dose acceptance criteria	 0.2 6.3	 0.1 6.3	 0.8 5
Control Rod Ejection Accident Secondary Side Release Pathway Dose acceptance criteria	 0.1 6.3	 0.1 6.3	 0.5 5
SGTR, Pre-existing spiking Dose acceptance criteria	0.3 25	0.1 25	3.9 5
SGTR, Accident-initiated spiking Dose acceptance criteria	0.2 2.5	0.1 2.5	1.1 5
LBLOCA, total Dose acceptance criteria	0.5 25	0.5 25	4.1 5
FHA Dose acceptance criteria	0.6 6.3	0.2 6.3	4.3 5
WGDT Rupture Dose acceptance criteria(1)	0.1 WB 0.5 WB	0.1 WB 0.5 WB	0.4 5
VCT Rupture Dose acceptance criteria(1)	0.1 WB 0.5 WB	0.1 WB 0.5 WB	0.6 5

(1) For the WGDT and VCT rupture accidents, the EAB and LPZ dose acceptance criteria are specified in the original licensing basis, Branch Technical Position 11-5 (reference 11), based on the earlier version of 10 CFR 20. Control room dose for these accidents is compared with the limits in GDC 19 (reference 15) and applicable standards in RG 1.183 (reference 1).

4.0 TECHNICAL ANALYSIS

The proposed changes to the KPS TS are discussed and evaluated below. Section 4.1 addresses the proposed changes associated with the revised RAA. These are changes that result from the revised RAA, exclusive of those required for establishing a CREH program. Section 4.2 addresses the proposed TS changes associated with implementation of the CREH program.

4.1 Technical Specification Changes Proposed due to Revised Radiological Accident Analysis

Based on revised inputs, assumptions and analysis, DEK is requesting changes to the TS to accommodate a proposed revision to the RAA. In conjunction with the changes necessary to accommodate the revised RAA, DEK is proposing changes that would adopt the following TSTFs:

- TSTF-312, Revision 1, "Administratively Control Containment Penetrations."
- TSTF-51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations."

4.1.1 Revise Specific Activity Limits

<u>TS 3.4.16, "RCS Specific Activity"</u>

The proposed amendment would reduce the current Reactor Coolant System (RCS) specific activity limits in TS 3.4.16 to values that are consistent with the revised RAA. The revised RAA assumes a DEI limit of ≤ 0.1 μCi/gm, a pre-existing iodine spike limit of ≤ 10 μCi/gm DEI, and a DEX limit of ≤ 16.4 μCi/gram.

TS 3.4.16 provides limits for the allowable concentration level of radionuclide's in the reactor coolant. The reactor coolant specific activity limits are established to minimize the dose consequences in the event of a main steam line break (MSLB) or steam generator tube rupture (SGTR) accident. TS 3.4.16 contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in Regulatory Guide (RG) 1.183[1].

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67, "Accident source term." Doses to control room operators must be limited per 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, "Control Room." The limits on reactor coolant specific activity

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[1] For the WGDT and VCT rupture accidents, the EAB and LPZ dose acceptance criteria are specified in the original licensing basis, Branch Technical Position 11-5 (reference 11), based on the earlier version of 10 CFR 20. Control room dose for these accidents is compared with the limits in GDC 19 (reference 15) and applicable standards in RG 1.183 (reference 1).

ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The revised RAA's in Attachment 4 assume the specific activity of the reactor coolant is at the proposed new limits and that, for specific accidents, an existing reactor coolant steam generator (SG) tube leakage rate of 150 gallons per day exists. The revised RAA concludes that the resulting dose consequences will be within the above limits. Therefore, the proposed changes to TS 3.4.16 are considered to be acceptable.

DEK has performed a calculation to determine the 1% failed fuel RCS coolant activity and the TS limits for the primary coolant for DEI-131 and RCS gross specific activity limit based on dose equivalent Xe-133 methodology. The following information is provided to demonstrate how DEI-131 and DEX-133 limits were developed.

Calculation of DE I-131

DE I-131 is determined in Table 1, from the 1% Failed Fuel inventory (KPS USAR Table D.4-1 values increased by 1.12/1.1 to allow core design uncertainty to increase from 5% to 10% as discussed in Attachment 4, Tables 3.4-4, 3.8-2, and 3.9-4 and Section 3.9.2.2), by multiplying the I-131 through I-135 isotopes by the ratio of its Thyroid CDE DCF divided by the I-131 Thyroid CDE DCF and then summing the results for each isotope. As shown in Table 1, there are 3.694 µCi/gm in 1% Failed Fuel. The 0.1 µCi/gm DE I-131 TS limit that is being proposed is then determined by multiplying the I-131 through I-135 isotopes in the 1% Failed Fuel by 0.1 µCi/gm and dividing by 3.694.

Table 4-1
RCS Coolant Concentrations for 0.1 µCi/gm Dose Equivalent I-131
(Proposed TS Limit - FGR 11 Thyroid CDE DCF)

Iodine Isotope	FGR 11 Table 2.1 Thyroid CDE DCF (Sv/Bq)	1% FF (µCi/gm)	1% FF DE I-131 (µCi/gm)	0.1 µCi/gm DE I-131 (µCi/gm)
I-131	2.92E-07	2.89E+00	2.89E+00	7.82E-02
I-132	1.74E-09	2.95E+00	1.76E-02	7.97E-02
I-133	4.86E-08	4.31E+00	7.18E-01	1.17E-01
I-134	2.88E-10	5.97E-01	5.89E-04	1.62E-02
I-135	8.46E-09	2.36E+00	6.85E-02	6.40E-02
total			3.694E+00	

Table 2 below, shows the DE I-131 calculation using FGR 11 CEDE DCFs instead of FGR 11 Thyroid CDE DCFs and results in a 2% reduction in source term.

Table 4-2
RCS Coolant Concentrations for 0.1 µCi/gm Dose Equivalent I-131
(FGR 11 CEDE DCF)

Iodine Isotope	FGR 11 Table 2.1 CEDE DCF (Sv/Bq)	1% FF (µCi/gm)	1% FF DE I-131 (µCi/gm)	0.1 µCi/gm DE I-131 (µCi/gm)
I-131	8.89E-09	2.89E+00	2.89E+00	7.64E-02
I-132	1.03E-10	2.95E+00	3.41E-02	7.79E-02
I-133	1.58E-09	4.31E+00	7.66E-01	1.14E-01
I-134	3.55E-11	5.97E-01	2.38E-03	1.58E-02
I-135	3.32E-10	2.36E+00	8.83E-02	6.25E-02
total			3.781E+00	

Hence it is conservative to use FGR 11 Thyroid CDE DCFs to determine DE I-131.

<u>Calculation of DE Xe-133</u>

Table 3 shows the calculation of the RCS activity limit for DE Xe-133 (µCi/gm) based on a source term proportional to 0.1 µCi/gm DE I-131 from Attachment 4, Table 3.4-1. DE Xe-133 is determined by multiplying the inventory of each Noble Gas isotope by the ratio of its FGR 12 DCF divided by the Xe-133 DCF and then summing the results for each isotope.

Table 4-3
DE Xe-133 for 0.1 µCi/gm DE I-131

	FGR 12 Table III.1 EDE DCF (Sv-m^3/Bq-sec)	NG Inventory α 0.1 µCi/gm DE I-131 (µCi/gm)	Xe-133 DCF (Sv-m^3/Bq-sec)	DE Xe-133 (µCi/gm)
Kr-85m	7.48E-15	4.76E-02	1.56E-15	2.28E-01
Kr-85	1.19E-16	2.37E-01	1.56E-15	1.81E-02
Kr-87	4.12E-14	3.11E-02	1.56E-15	8.22E-01
Kr-88	1.02E-13	9.03E-02	1.56E-15	5.91E+00
Xe-131m	3.89E-16	8.37E-02	1.56E-15	2.09E-02
Xe-133m	1.37E-15	9.47E-02	1.56E-15	8.32E-02
Xe-133	1.56E-15	6.67E+00	1.56E-15	6.67E+00
Xe-135m	2.04E-14	1.38E-02	1.56E-15	1.80E-01
Xe-135	1.19E-14	2.39E-01	1.56E-15	1.83E+00
Xe-138	5.77E-14	1.73E-02	1.56E-15	6.40E-01
DE Xe-133 (µCi/gm)				**16.4**

TS 3.7.16, “Secondary Specific Activity”

In conjunction with a proposed decrease to the reactor coolant specific activity limits, a reduction of the secondary coolant specific activity limit is also proposed. This amendment proposes to revise TS 3.7.16, “Secondary Specific Activity,” to incorporate a new secondary coolant specific activity limit. Specifically, the secondary coolant specific activity limit in LCO 3.7.16 and SR 3.7.16.1 would be reduced from ≤ 0.10 μCi/gm DEI to ≤ 0.05 μCi/gm DEI. The allowable activity levels are intended to ensure that offsite and control room doses meet the applicable acceptance criteria in RG 1.183.

Limiting secondary coolant specific activity during power operation minimizes releases to the environment during normal operation, anticipated operational occurrences, and accidents. The limits on secondary coolant system specific activity ensure that the analyzed post-accident dose consequences of design basis accidents are below the limits in 10 CFR 50, Appendix A, GDC 19, “Control Room” and 10 CFR 50.67, “Accident source term.” The revised RAA in Attachment 4 assumes a secondary coolant specific activity limit of ≤ 0.05 μCi/gm DEI and concludes that the resulting dose consequences will be within the above limits. Therefore, the proposed changes to TS 3.7.16 are considered to be acceptable.

4.1.2 Adoption of TSTF-51

DEK is proposing to adopt TSTF-51, “Revise containment requirements during handling of irradiated fuel and core alterations.” TSTF-51 permits the removal of TS requirements for certain ESF features (e.g., primary/secondary containment isolation capability) to be OPERABLE after sufficient radioactive decay of the nuclear fuel has occurred to ensure off-site post-accident doses remain below 10 CFR 50.67 limits. Fuel movement would still be allowed prior to sufficient radioactive decay occurring, but only with the appropriate ESF systems OPERABLE.

TSTF-51 also allows flexibility in moving personnel and equipment into and out of the containment, and in performing work affecting containment operability, during the movement of irradiated fuel.

Following a reactor shutdown, radioactive decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Adoption of TSTF-51 is based on performing a radiological analysis which assumes a longer decay period in order to take advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. Following sufficient radioactive decay occurring, the primary success path for mitigating the radiological effects of a fuel handling accident no longer includes the functioning of the active containment systems. Therefore, the OPERABILITY requirements of the TS are being modified to reflect that water level and decay time are the primary success paths for mitigating a fuel handling accident (which meets 10 CFR 50.36 (c)(2)(ii)(C), “Criterion 3”).

KPS TS 3.9.5, “Refueling Cavity Water Level,” allows movement of irradiated fuel assemblies within containment only if water level in the refueling cavity is greater than or equal to 23 feet above the top of the reactor vessel flange. Therefore, implementation of TSTF-51 only affects containment requirements during periods of relatively low shutdown risk during refueling outages.

Recently Irradiated Fuel (RIF) is defined as; “fuel that has occupied part of a critical reactor core within the preceding “X” days (or hours)”. “X” is a site specific number based on meeting the limits for radiological exposure in 10 CFR 50.67. This definition is used in development of the TS changes described in Section 2.1.2. The value of “X” is derived by assuming that a FHA occurs and no mitigative features are in place to assist in exposure reduction (i.e. containment, fission product removal system, etc.).

The current and proposed RAA’s both assume 100 hours of decay time has occurred prior to movement of irradiated fuel assemblies. After 100 hours of decay time, the FHA radiological accident analysis shows acceptable dose results at the EAB and LPZ without crediting containment and its associated systems. The KPS Technical Requirements Manual (TRM) Section 8.9.3, “Decay Time,” requires that the reactor be subcritical for at least 100 hours before irradiated fuel can be moved within the reactor vessel. The 100 hour limit in the TRM is the same amount of time as assumed in the dose calculations for the FHA. Therefore, TRM 8.9.3 ensures this radiological analysis assumption is implemented. After 375 hours of decay time, analysis using the assumptions of the FHA shows acceptable dose results for the control room occupants without crediting any control room emergency ventilation or operator action. Irradiated fuel movement would still be allowed after 100 hours and prior to 375 hours of decay time occurring provided appropriate systems are OPERABLE. Details concerning development of the value for defining recently irradiated fuel are found in Attachment 4 (see Section 3.3.5.4, “Recently Irradiated Fuel Determination”).

Based on the above, the definition of recently irradiated fuel for KPS is proposed to be; “fuel that has occupied part of a critical reactor core within the previous 375 hours.” This definition will be added to the TS Bases for the TS listed in Section 2.1.2, consistent with TSTF-51.

In order to adopt TSTF-51, licensees must make a commitment consistent with draft NUMARC 93-01, Revision 3, Section 11.2.6, “Safety Assessment for Removal of Equipment from Service During Shutdown Conditions,” subheading, "Containment – Primary (PWR)/Secondary (BWR)." Therefore, DEK makes the following commitment:

“The following guidelines will be included in the assessment of systems removed from service during movement of recently irradiated fuel:

a. During fuel handling of recently irradiated fuel, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) will be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

b. *A single normal or contingency method to promptly close primary or secondary containment penetrations will be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.*

The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

Based on the discussion above, DEK believes the changes proposed to adopt TSTF-51 are acceptable.

4.1.3 Revise TS 3.9.6, Containment Penetrations

DEK is proposing changes to TS 3.9.6, "Containment Penetrations," to provide the flexibility to open containment penetration flow paths under administrative controls during refueling outage periods. The proposed changes to TS 3.9.6 are discussed in Section 2.1.3 and shown in Attachment 2.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within 10 CFR 50.67 limits. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions. A description of the KPS containment is provided in Section 3.1 of this Attachment.

TS 3.9.6 is a Refueling Operations TS and is currently applicable only during movement of irradiated fuel assemblies within containment. During refueling operations, the potential for containment pressurization as a result of an accident is not likely. Therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO 3.9.6 requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since the potential for containment pressurization as a result of an accident is not likely, containment system "integrity" is not required during refueling operations.

The equipment hatch and the containment personnel air locks are part of the containment pressure boundary and their associated requirements are included in LCO 3.9.6.a and LCO 3.9.6.b, respectively. All other containment penetration flow paths are addressed in LCO 3.9.6.c and are currently required to be closed during refueling operations or capable of being closed by an operable Containment Purge and Vent Isolation System.

The proposed changes to LCO 3.9.6.a would allow the containment equipment hatch to be open and capable of being closed while moving recently irradiated fuel assemblies within containment. In addition a Note would be added allowing containment penetration flow paths to be open under administrative controls. The proposed changes to LCO 3.9.6.c would modify the requirements related to the status of containment penetration flow paths during refueling operations.

<u>Incorporate the term “recently” into the APPLICABILITY and Required Action A.1 of TS 3.9.6</u>

Consistent with the adoption of TSTF-51, this change adds the term “recently” to the APPLICABILITY and Required Action A.1 of TS 3.9.6 so that they will state: During [Suspend] movement of [recently] irradiated fuel assemblies within containment.” This change is consistent with adoption of TSTF-51 and is discussed in Section 4.1.2 above.

<u>Allow penetration flow paths providing direct access from the containment to outside atmosphere to be opened under administrative controls</u>

DEK is proposing to adopt TSTF-312, “Administratively Control Containment Penetrations.” Consistent with adoption of TSTF-312, a new Note is added which modifies LCO 3.9.6. The new Note would allow containment penetration flow paths providing direct access from the containment to outside atmosphere to be open under administrative control during movement of recently irradiated fuel assemblies within containment.

In accordance with TSTF-312, the allowance to have containment penetration flow paths with direct access from the containment atmosphere to the outside atmosphere open during movement of recently irradiated fuel within containment is based on:

1. Confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences, and;
2. A commitment to implement acceptable administrative procedures that ensure, in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences), that the open penetration(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

This proposed change is based on the revised RAA provided in Attachment 4. The revised RAA assumes a decay period sufficient to take advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. Following sufficient decay time (100 hours), the primary success path for mitigating a fuel handling accident does not require active containment isolation systems to function (i.e., containment penetrations have been modeled as open in the revised fuel handling accident analysis). The resulting control room, EAB, and LPZ doses are less than the acceptance criteria of 10 CFR 50.67 (as modified by RG 1.183). Therefore, the containment penetrations may be open during movement of recently irradiated fuel assemblies within containment because the confirmatory dose calculations for the fuel handling accident indicate acceptable radiological consequences. Consistent with item

1 above, DEK has developed confirmatory dose calculations for a fuel handling accident which indicate acceptable radiological consequences.

Consistent with item 2 above, DEK provides the following commitment:

> *"DEK will provide the necessary administrative controls to ensure that in the event of a fuel handling accident inside containment, any open containment penetration flow paths can and will be promptly closed."*

When penetration flow paths are open during movement of recently irradiated fuel within containment, the following administrative controls will be in place. (It is noted that similar administrative control requirements are currently in place for the containment air lock doors and will also be provided for the equipment hatch).

1. Appropriate personnel are aware of the open status of the containment penetration flow path during movement of recently irradiated fuel assemblies within containment;
2. Specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident inside containment, and;
3. Any obstruction(s) (e.g., cables and hoses) that could prevent closure of any containment penetration can be quickly removed.

The time for closure of the penetration flow paths following a FHA is 30 minutes or less. This closure time is consistent with the guidance of RG 1.183 for such operations. However, if it is determined that closure of any containment penetrations would represent a significant radiological hazard to the personnel involved; the decision may be made to forgo the closure of the affected penetration(s).

<u>Allow containment equipment hatch to be open and capable of being closed during movement of recently irradiated fuel assemblies within containment</u>

Currently, LCO 3.9.6.a requires the equipment hatch to be closed and held in place by four bolts during movement of irradiated fuel assemblies within containment. This proposed change would modify TS 3.9.6.a to allow the containment equipment hatch to be open during movement of recently irradiated fuel assemblies when measures are in place that ensure the capability to close the equipment hatch in the event of a fuel handling accident.

Unlike the containment penetration flow paths and personnel air lock, the NRC has not specifically endorsed a TSTF for permitting the equipment hatch to be open during handling of recently irradiated fuel inside containment. TSTF-312 could include the equipment hatch, as it meets the definition of a containment penetration flow path "with direct access from the containment atmosphere to the outside atmosphere." Therefore, DEK proposes that this change for the equipment hatch be subject to the same requirements as similar changes related to penetration flow paths and personnel air locks.

Specifically, consistent with TSTF-312, the proposed change to allow the containment equipment hatch to be open to the outside atmosphere during movement of recently irradiated fuel assemblies within containment is based on:

1. Confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences, and;
2. A commitment to implement acceptable administrative procedures that ensure, in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences), that the equipment hatch can and will be promptly closed following containment evacuation. The time to close the equipment hatch shall be included in the confirmatory dose calculations.

This proposed change is based on the revised RAA provided in Attachment 4. The revised RAA assumes a decay period sufficient to take advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. Following sufficient decay time (100 hours), the primary success path for mitigating a fuel handling accident does not require active containment isolation systems to function (i.e., the equipment hatch has been modeled as open in the revised fuel handling accident analysis). The resulting Control Room, EAB, and LPZ doses are less than the acceptance criteria of 10 CFR 50.67 (as modified by RG 1.183). Therefore, the equipment hatch may be open during movement of recently irradiated fuel assemblies within containment because the confirmatory dose calculations for the fuel handling accident indicate acceptable radiological consequences. Consistent with item 1 above, DEK has developed confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences.

Consistent with item 2 above, DEK provides the following commitment:

> *"DEK will provide the necessary administrative procedures to ensure that in the event of a fuel handling accident inside containment, the open equipment hatch can and will be promptly closed following containment evacuation."*

When the equipment hatch is open during movement of recently irradiated fuel within containment the following administrative controls will be in place. (It is noted that similar administrative controls are currently in place for the containment air lock doors and will also be provided for other containment penetrations, as previously discussed):

1. Appropriate personnel are aware that the equipment hatch is open;
2. A specified individual(s) is designated and available to close the equipment hatch following a required evacuation of containment, and;
3. Any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch can be quickly removed.

The estimated time to close the equipment hatch following evacuation of containment after a FHA is about 45 minutes. This closure time is an exception to the 30 minute closure time recommended in RG 1.183 for such operations. However, DEK proposes that the difference between the 30 minute closure time recommended by RG 1.183 and the estimated 45 minute closure time for the equipment hatch is acceptable because the RAAs demonstrate that the offsite dose limits are not exceeded without closure of containment.

Therefore, even though the containment fission product control function of the equipment hatch is not required to meet 10 CFR 50.67 dose limits in this case, DEK will provide the necessary administrative controls to ensure that, in the event of a fuel handling accident inside containment, the equipment hatch can and will be promptly closed following containment evacuation. However, if it is determined that closure of the containment hatch would represent a significant radiological hazard to the personnel involved; the decision may be made to forgo the closure of the containment hatch.

Since the calculated dose consequences, assuming the equipment hatch remains open for the duration of an FHA, are well below 10 CFR 50.67 acceptance criteria, DEK considers this change acceptable.

4.1.4 Revise TS 1.1 Definition of Dose Equivalent I-131

Section 2.1.4 proposes to revise the current definition of Dose Equivalent I -131 (DEI) in TS Section 1.1, consistent with the definition contained in TSTF-490 (reference 8). The new definition references Table 2.1 of EPA Federal Guidance Report No. 11 (FGR No. 11) (reference 14) as the source of thyroid CDE dose conversion factors based on the use of this conversion factor in the enclosed RAA (see Attachment 4). The dose conversion factors for inhalation used in the revised RAA are from Table 2.1 of FGR No. 11. Previously, the DEI conversion factors were based on ICRP 30. Essentially, the values in FGR No. 11 are derived from ICRP 30. With respect to ICRP 30, FGR No. 11 states:

> *"The ALI (Annual Limit on Intake) and DAC (Derived Air Concentration) values tabulated in FGR 11 are identical to those of ICRP 30, except for the isotopes of Np, Pu, Am, Cm, Bk, Cf, Es, Fm, and Md."*

In addition, NRC RIS 2001-19 (reference 28) states:

> *"The NRC staff considers thyroid dose conversion factors based on ICRP-30, such as those tabulated in Federal Guidance Report 11, to be an acceptable change in methodology that does not warrant prior review."*

Using Table 2.1 of FGR No. 11 for dose conversion factors is the most appropriate selection for plants using AST methodology. RG 1.183, Section 4.1.2 prescribes the use of FGR No. 11. It is appropriate for plants using the AST methodology to incorporate a definition of DEI based on the CDE dose conversion factors rather than thyroid dose conversion factors. This is because AST reported doses are not based on thyroid and whole body doses, but are based on Total Effective Dose Equivalent (TEDE).

Dose Equivalent I-131 is that concentration of I-131 (μCi/gm) 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 dose conversion factors (DCFs) used to determine dose from iodine are from Federal Guidance Report No. 11 (FGR-11), Table 2.1 committed effective dose equivalent (CEDE) and the calculation of the Dose Equivalent I-131 from proposed technical specification surveillance are from FGR-11 Table 2.1 Thyroid Committed Dose Equivalent (CDE). The acceptability for the pre-accident and concurrent iodine spike source terms to be based on FGR-11 Thyroid CDE DCFs, and the doses to be calculated using FGR-11 CEDE DCFs, is being submitted for NRC staff approval in this amendment request (see Attachment 4).

DEK has analyzed the consistency of the proposed definition for Dose Equivalent I-131, and its surveillance limits, and the DCFs used for the determination of Dose Equivalent I-131 surveillance limits. The site-specific limits for Dose Equivalent I-131, the DCFs, and the RCS radioistopic concentrations are consistent with the proposed design basis dose analyses (Steam Generator Tube Rupture (SGTR) and Main Steam Line Break (MSLB) for KPS. Other DCFs may be used in the analysis of other events, but those DCFs are not used to calculate the limits in LCO 3.4.16, "RCS Specific Activity."

In addition, RG 1.183 requires that the pre-accident and concurrent iodine spikes used in design basis accident (DBA) analysis be based on the maximum values permitted by the Technical Specifications, which are proposed to be 10 μCi/gm and 0.1 μCi/gm, respectively, for KPS. The KPS MSLB and SGTR accidents are analyzed using the maximum allowed reactor coolant system activity. CEDE Dose conversion factors from FGR-11 are used to calculate the TEDE consequences described using the guidance from RG 1.183, while the 0.1 μCi/gm Dose Equivalent I-131 inventory is calculated using FGR-11 Thyroid CDE DCFs. FGR-11 Thyroid CDE DCFs result in a slightly higher total allowable iodine inventory in the RCS than would be attainable using FGR-11 CEDE DCFs, which results is slightly higher dose consequences for the SGTR and MSLB accident analyses.

Therefore, use of EPA FGR No. 11, Table 2.1 for CDE dose conversion factors is considered acceptable.

4.2 Technical Specification Changes Proposed to Establish Control Room Envelope Habitability Requirements

The purpose of the following changes is to incorporate specific requirements into the KPS TS that are applicable to Control Room Envelope Habitability (CREH). The proposed changes are based on TSTF-448. It is noted that the KPS control room is a neutral-pressure control room, (i.e., it is not intentionally pressurized during accident conditions). This results in some differences between TSTF-448 and the proposed KPS OL and TS changes. The proposed changes are consistent with the revised RAA (see Attachment 4).

Regarding adoption of TSTF-448, DEK has reviewed the safety evaluation dated January 17, 2007, as part of the CLIIP. This review included a review of the NRC staff evaluation, as well as the supporting information provided to support TSTF-448. DEK has concluded that the justifications presented in the TSTF proposal, and the safety evaluation prepared by the NRC staff, are applicable to KPS and justify incorporation of the proposed changes to the TS. Since TSTF-448 is applicable to KPS, only deviations from the TSTF will be justified in the evaluations provided below.

4.2.1 Add Control Room Envelope Habitability Program

DEK is proposing to add TS 5.5.17, “Control Room Envelope Habitability Program,” as prescribed by TSTF-448. The wording of the proposed new TS 5.5.17 is shown in Attachment 2. The proposed addition of TS 5.5.17 is modeled after the TSTF-448 CLIIP. Deviations and bracketed information from the CLIIP are discussed below.

1. The CLIIP for TSTF-448 reference to “TS 5.5.18” was changed to “TS 5.5.17,” which corresponds to the proposed numbering of the KPS TS CREH Program.

2. The CLIIP for TSTF-448 requires licensees to select either bracketed item out of the following sentence.

   “The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of [5 rem whole body or its equivalent to any part of the body] [5 rem total effective dose equivalent (TEDE)] for the duration of the accident.”

   DEK has selected *[5 rem total effective dose equivalent (TEDE)].* This selection is appropriate because KPS has adopted the AST methodology.

3. The CLIIP for TSTF-448 requires adoption of NUREG-1431, Section 5.5.18.d which states the following:

   “Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a

Frequency of [18] months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary."

The KPS control room is a neutral pressure (i.e., non-pressurized) control room. The control room ventilation system is designed to maintain the control room at a neutral pressure. Because the KPS control room is a neutral pressure control room, the results of measuring differential pressure between the control room and adjacent areas are subject to much variation. This variation may be caused by ventilation system configurations, location and sensitivity of instrumentation, and the effects of weather conditions and building structures on intake air. Furthermore, in GL 2003-01 the NRC also called into question the usefulness of differential pressure measurements of CREs in general.

A review of the pressure data taken during the tracer gas testing performed on the KPS CRE in December 2004 concluded that a pressurization (differential pressure) test requirement for the CRE boundary would not yield useful data regarding the leak tightness of the KPS neutral pressure control room. The test report for the KPS CRE tracer gas test indicated that some adjacent areas were at positive pressure, and other adjacent areas were at negative pressure, with respect to the CRE.

During discussions between the NRC staff and the industry, it was recognized that facilities with non-pressurized CREs may not be able to conduct meaningful differential pressure tests. Nevertheless, the NRC staff believed that all plants requesting adoption of TSTF-448 should include in their request a method to collect data that will serve as input to a periodic assessment of the CRE boundary. The use of programs to verify the integrity of the CRE boundary, including the use of the corrective action program and trending of relevant information as part of the assessment program, will provide additional assurance that significant degradation of the CRE boundary will not go undetected between CRE inleakage determinations (see reference 13, page 10).

Rather than performing periodic differential pressurization tests, DEK conducts preventative maintenance (PM) and surveillance tests (STs) that provide reasonable assurance that the CRE boundary is maintained in a manner that will provide adequate protection for the operators. The data from these PMs and STs will be used as a subjective means of assessing the condition of the CRE between the quantitative in-leakage tracer gas tests. A description of these PMs and STs is provided below.

a. **Damper Maintenance (PM)** - Once every two cycle inspection of control room ventilation damper mechanical components and internal blade seals, with parts replaced as necessary.

b. **Control Room Air Conditioning Mechanical Inspection and Maintenance (PM)** - Once per year inspection and replacement of control room ventilation boundary components including:

i. Door inspections - Inspection and repair/replacement of weather strip seals of the control room envelope doors.

ii. Cable tray penetration inspections - Inspection and repair (as necessary) of the relay room electrical cable tray penetrations.

c. **Penetration Fire Barrier Inspection (PM)** - Once per cycle inspection/repair (as necessary) of fire and steam exclusion barrier penetrations, including those associated with the CRE.

d. **Post Accident Recirculation Test (ST)** - Monthly functional test of the emergency ventilation filter components.

e. **Control Room Post Accident Recirculation Train Operability Testing (ST)** - Monthly test of the CRPAR system. Each Train is operated for at least 15 minutes.

f. **Control Room Post Accident Train Recirculation Filter Testing (ST)** – At least once per 18 month test of HEPA filters and charcoal adsorber banks. Fan flows are obtained during this test. Fan flows are maintained within design flow rate limits during this test. The charcoal adsorber flow rates are maintained within the TS limits. Maintaining proper flow rates reduces the possibility of control room pressure changes, which may affect CRE unfiltered in-leakage.

g. **Auxiliary Building Special Ventilation Operability Test (ST)** - Monthly verification of some of the CRE boundary dampers to close. These dampers are also steam exclusion dampers.

h. **Barrier Control Procedure** - Provides instructions for managing and controlling the integrity of the CRE. This includes permitting and managing openings in the CRE, including total opening size. This procedure also provides instructions for managing barrier impairments and logging TS requirements for openings in the CRE.

These activities provide reasonable assurance that the KPS CRE boundary will perform its safety function and are considered an acceptable alternative to performing periodic pressurization tests and trending the test data. Any criteria not met while performing these activities are documented and resolved in accordance with the corrective action program. Appropriate actions are identified and implemented to address identified non-conforming or degraded conditions and assess the impact on the CRE boundary.

Therefore, DEK proposes an exception to the CLIPP for TSTF-448 consistent with the KPS plant specific design and the discussion above. DEK proposes the following wording in place of the wording above:

"Licensee controlled programs will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will

be entered into the corrective action process and shall be trended and used as part of a 36-month assessment program for the CRE boundary in accordance with TS 5.5.17.c.2."

This wording is incorporated as TS 5.5.17 in the marked up pages provided in Attachment 2.

4. The CLIIP for TSTF-448 requires adoption of NUREG-1431, Section 5.5.18.f, which states the following.

   "The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively."

   As discussed above, since KPS has a neutral pressure control room, measurement of CRE pressure is not a reliable measure of CRE boundary integrity. Therefore, DEK proposes removal of the phrase, *"and measuring CRE pressure"* from the CLIIP language above.

   The final proposed wording is provided as TS 5.5.17.f in the marked up pages provided in Attachment 2.

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

DEK is proposing to modify TS 3.7.10, "Control Room Post Accident Recirculation (CRPAR) System," consistent with adoption of TSTF-448. The proposed changes to TS 3.7.10 are shown in Attachment 2 and are discussed below. The CRPAR system is the KPS system equivalent of the Control Room Emergency Filtration System (CREFS) as discussed in TSTF-448. A description of the CRPAR system is provided in Section 3.1.1 of this Attachment.

The proposed changes to TS 3.7.10 follow the model TS provided in TSTF-448. Each of the proposed changes is listed below. The proposed changes that are consistent with TSTF-448 are designated as such. In addition, the changes that are not within the scope of TSTF 448 are also identified and discussed below. The proposed changes to TS 3.7.10 also include adoption of TSTF-51 (use of the term "recently irradiated fuel") as previously discussed and evaluated in Section 4.1.2 of this Attachment.

1. The existing NOTE in LCO 3.7.10 would be modified to change the current wording from; "*The control room boundary may be opened intermittently under administrative control"* to "*The control room envelope (CRE) boundary may be opened intermittently under administrative control*." This change follows the model TS provided in TSTF-448.

2. A new NOTE would be added to LCO 3.7.10 which states; “The CRE shall be isolated during movement of recently irradiated fuel assemblies.” Currently, LCO 3.7.10 requires that two CRPAR trains shall be operable during Modes 1-6 and during movement of irradiated fuel assemblies within containment. To ensure control room doses following a FHA remain below applicable acceptance criteria, the revised RAA in Attachment 4 assumes the control room is isolated at the initiation of a FHA. Pre-isolation of the control room minimizes infiltration of radioactive materials into the CRE prior to initiation of the CRPAR system in the emergency mode and ensures dose to CRE occupant’s remains within applicable limits.

3. The current APPLICABILITY for LCO 3.7.10 is Modes 1-6, and during movement of irradiated fuel assemblies. The APPLICABILITY for LCO 3.7.10 would be changed to Modes 1-4, and during movement of recently irradiated fuel assemblies. Consistent with this change, TS 3.7.10 Condition D and Condition E are also modified by removing Mode 5 and 6 applicability.

   Currently TS 3.7.10 requires the CRPAR system to be operable in MODE 5 and 6. The current TS 3.7.10 Bases state that in Modes 5 and 6, the CRPAR system must be operable to; 1) control operator exposure during and following a DBA, and 2) to cope with the release from a rupture of an inside waste gas tank. DEK proposes to delete the Mode 5 and 6 applicability of TS 3.7.10 and revise the TS Bases consistent with this proposed change.

   The KPS Waste Gas Decay Tanks (WGDT) (USAR Chapter 11.1) (reference 19) and Volume Control Tank (VCT) (USAR Chapter 9.2) are located inside the auxiliary building, where radioactive gases are collected and filtered prior to release.

   The WGDT failure and VCT rupture (atmospheric release) radiological analyses are being revised to reflect revised X/Q values as discussed in Attachment 4 of this application. For the WGDT and VCT rupture accidents, the EAB and LPZ dose acceptance criteria are specified in the original licensing basis, Branch Technical Position 11-5 (Reference 11), based on the earlier version of 10 CFR 20. Control room dose for these accidents is compared with the limits in GDC 19 (reference 15) and applicable standards in RG 1.183. The revised WGDT and VCT analyses demonstrate acceptable dose to control room operators without credit for the control room emergency ventilation filtration or CRE isolation. These analyses also demonstrate acceptable dose at the EAB and LPZ under Branch Technical Position (BTP) ETSB 11-5, Revision 0.

   The only other design basis radiological accident postulated to occur when the plant is in Modes 5 and 6 is the Fuel Handling Accident. A FHA is postulated to occur only during movement of irradiated fuel and TS 3.7.10 will continue to be applicable during the movement of recently irradiated fuel, as discussed in item 2 above and in Section 4.1.2.

4. TS 3.7.10, Condition A wording would be modified from "*One CRPAR Train inoperable*" to "*One CRPAR Train inoperable for reasons other than Condition B.*" This change follows the model TS provided in TSTF-448.

5. TS 3.7.10, Condition B currently provides a Required Action when two CRPAR trains are inoperable due to an inoperable CRE boundary in Modes 1-4. The current Required Action B.1 is to restore the CRE boundary within 24 hours. Condition B and its associated Required Action and Completion Time would be replaced with a new Condition B. The new Condition B would provide required actions and completion times when one or more CRPAR trains are inoperable due to an inoperable CRE boundary in Modes 1-4.

   The new Condition B would provide three new Required Actions when one or more CRPAR trains are inoperable due to an inoperable CRE boundary in Modes 1-4. The three new Required Actions (B.1, B.2, and B.3) would require; (B.1) immediate initiation of action to implement mitigating actions; (B.2) within 24 hours, verification that mitigating actions ensure CRE occupant exposures to radiological, chemical and smoke hazards will not exceed limits and; (B.3) restoration of the CRE boundary to operable status within 90 days.

   These proposed changes follow the model TS provided in TSTF-448.

6. TS 3.7.10, Condition E currently requires immediate suspension of movement of irradiated fuel assemblies when two CRPAR trains are inoperable in Modes 5 and 6 and during movement of irradiated fuel assemblies. The current Condition E would be modified by deleting Mode 5 and 6 applicability (see item 3 above) and adding the word "recently" so that the resulting condition would state; "Two CRPAR trains inoperable during movement of recently irradiated fuel assemblies." In addition, Condition E would be expanded to include situations where the Required Actions and associated Completion Times of Condition B are not met during movement of recently irradiated fuel assemblies. The Required Action and Completion Time of Condition E would remain unchanged except for incorporation of the term "recently" into Required Action E.1, consistent with adoption of TSTF-51. .

   The deletion of Mode 5 and 6 applicability is addressed in item 3 above. The other changes follow the model TS provided in TSTF-448.

7. A new Surveillance Requirement would be added. New SR 3.7.10.4 would require unfiltered air inleakage testing of the Control Room Envelope in accordance with the proposed new Control Room Envelope Habitability Program. This change follows the model TS provided in TSTF-448. A discussion of the proposed new Control Room Envelope Habitability Program is provided in Section 4.2.1.

4.2.3 Delete TS Table 3.3.7-1, Function 2, Control Room Vent Monitor

Section 2.2.3 proposes deletion of TS Table 3.3.7-1, Function 2, "Control Room Vent Monitor." Consistent with deletion of Table 3.3.7.1, Function 2, the portion of TS 3.3.7

Condition B which states "OR Control Room Vent Radiation Monitor inoperable," would be deleted. In addition, SR 3.3.7.1, SR 3.3.7.2, and 3.3.7.4 would be deleted since these SRs are solely applicable to the Control Room Vent Radiation Monitor (Function 2 in Table 3.3.7-1). These changes are based on the revised RAA in Attachment 4.

The control room ventilation radiation monitor consists of a single radiation monitor (R-23) located on the common discharge of the outlet of the 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 fans by manual switches in the control room. The CRPAR system is also actuated by a safety injection signal. A detailed discussion concerning the CRPAR system and R-23 is provided in Section 3.1.1 of this Attachment.

DEK is proposing to delete R-23 as a required channel for CRPAR initiation. The revised RAA in Attachment 4 does not rely on or credit radiation monitor R-23 to isolate the control room during radiological events. In Section 4.2.2 above, a new Note is being added to TS 3.7.10 which would require the control room to be isolated prior to movement of recently irradiated fuel. This new Note is consistent with the revised RAA, which assumes the control room is isolated prior to moving recently irradiated fuel.

Thus, reliance on R-23 to isolate the control room in the event of a FHA is no longer necessary. For other DBAs, the revised RAA assumes reasonable operator actions or a safety injection signal will perform the necessary control room isolation function and maintain doses within acceptable limits.

Specifically, in accordance with the revised RAA, DEK is proposing two manual actions to ensure post-accident control room dose is maintained within limits. The revised RAA indicates that manual actions are required to limit consequences of the FHA and LRA events. The proposed manual actions are as follows:

1. The revised RAA credits manual operator action to isolate the control room within one hour after initiation of a Locked Rotor Accident (LRA). This manual action is required to compensate for the proposed TS changes that would discontinue credit for control room auto-isolation using a high radiation signal from R-23.
2. The revised RAA assumes the CRE is isolated prior to movement of recently irradiated fuel assemblies (per new Note added to TS 3.7.10). In addition, the revised RAA credits manual initiation of the Control Room Post Accident Recirculation (CRPAR) system within 20 minutes of occurrence of a FHA.

An evaluation of the acceptability of the proposed new manual actions is provided in Attachment 5.

The equipment necessary to initiate control room isolation and starting of the CRPAR trains is tested monthly as part of SR 3.7.10.1, which requires operation of each CRPAR train for greater than or equal to 15 minutes on a 31-day frequency. Because this equipment consists of pushbuttons and switches, no checks or calibrations are required.

Deletion of SR 3.3.7.1, SR 3.3.7.2, and 3.3.7.4 is consistent with the deletion of R-23 from the TS because these three SR's apply only to R-23 and no other plant equipment.

Therefore, based on the results of the revised RAA, R-23 actuation is no longer included or credited in primary success path to mitigate the consequences of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Based on this conclusion, R-23 no longer meets the criteria provided in 10 CFR 50.36(c) for inclusion in the TS. DEK intends to maintain R-23 in service and functional as a defense-in-depth measure. DEK will relocate the TS requirements related to R-23 to the KPS Technical Requirements Manual and maintain a description of R-23 in the USAR. Control of future changes to the relocated requirements and the instrumentation itself will be in accordance with 10 CFR 50.59 requirements.

4.2.4 License Condition for Implementation of TSTF-448 Requirements

Section 2.2.4 proposes the addition of a new license condition to the KPS OL associated with adoption of TSTF-448. This license condition is identical (with one exception discussed below) to the license condition found in the model for adoption of TSTF-448, as amended by NRC letter dated February 2, 2007 (reference 16).

An exception to TSTF-448 is that Item c of the model license condition has not been included in the proposed license condition for KPS. Item c of the model license condition provides scheduling requirements for the performance of the periodic measurement of CRE pressure after implementation of TSTF-448. As previously discussed in Section 4.2.1, KPS has a neutral pressure control room and therefore, a meaningful control room pressurization test is not possible. Instead, those facilities with neutral pressure control rooms have worked with the NRC staff and developed an acceptable alternative to performing control room pressurization tests. This alternative is included in proposed TS 5.5.17 and states:

> *"Licensee controlled programs will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the 36-month assessment of the CRE boundary in accordance with TS 5.5.17.c.2."*

Therefore, consistent with the KPS design and adoption of TS 5.5.17, "Control Room Envelope Habitability Program," DEK has not provided a license condition associated with periodic measurement of CRE pressure.

4.3 Conclusions

The proposed amendment would revise the KPS OL, TS, and USAR to incorporate changes resulting from a revised radiological accident analysis (RAA) and changes to implement a commitment relating to Control Room Envelope Habitability.

A revised RAA is included in Attachment 4. The RAA has been performed in accordance with RG 1.183, and concludes the plant meets the dose consequences acceptance criteria of 10 CFR 50.67.[2] 10 CFR 50.67(b)(2) states that the analysis must demonstrate with reasonable assurance that:

- *An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE).*
- *An individual located at any point on the outer boundary of the low-population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE.*
- *Adequate radiation protection is provided to permit access to and occupancy of the control room (CR) under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.*

The revised RAA contains revised assumptions and requirements for meeting the acceptance criteria of 10 CFR 50.67 described above. The results of the revised RAA provide reasonable assurance of meeting these acceptance criteria. Therefore, the changes proposed as a result of the revised RAA are considered acceptable.

In addition, in accordance with commitments made in response to GL 2003-01, DEK is adopting TSTF-448 by incorporating applicable changes into the KPS TS. This requires adoption of a new OL condition. These changes are also based on and consistent with the revised RAA. Adoption of other TSTFs has been proposed that support the control room envelope habitability requirements of TSTF-448 and the revised RAA. These proposed changes have been evaluated above and are also considered acceptable.

[2] For the WGDT and VCT rupture accidents, the EAB and LPZ dose acceptance criteria are specified in the original licensing basis, Branch Technical Position 11-5 (Reference 11), based on the earlier version of 10 CFR 20. Control room dose for these accidents is compared with the limits in GDC 19 (reference 15) and applicable standards in RG 1.183 (Reference 1).

5.0 REGULATORY SAFETY ANALYSIS

Regarding proposed changes in this application made in accordance with TSTF-448, DEK makes the following statement:

> DEK has reviewed the proposed No Significant Hazards Consideration Determination (NSHCD) published in the Federal Register as part of the CLIIP. DEK has concluded that the proposed NSHCD presented in the Federal Register notice is applicable to KPS and is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

5.1 Significant Hazards Consideration

DEK has evaluated the remainder of the LAR as to whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

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

Response: No

The proposed amendment would incorporate a revised radiological accident analyses (RAA) for Kewaunee Power Station (KPS).

The full implementation of revised alternative source term (AST) assumptions has been evaluated in a revision to the RAA of the following KPS design basis accidents (DBAs):

- Main Steam Line Break Accident
- Locked Rotor Accident
- Rod Ejection Accident
- Steam Generator Tube Rupture Accident
- Loss-of-Coolant Accident
- Waste Gas Decay Tank Rupture
- Volume Control Tank Rupture
- Fuel Handling Accident

Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory criteria established by the NRC for use with the AST. This criteria is specified in 10 CFR 50.67 and associated RG 1.183. Therefore, the consequences of an accident previously evaluated are not significantly increased by the proposed changes.

The equipment affected by the proposed amendment is mitigative in nature, and relied upon after an accident has been initiated. Modification of the AST assumptions does not require any physical changes to the plant design or plant equipment (systems, structures, or components). While the operation of various systems would change as a result of the proposed amendment, these systems are not accident initiators.

Revision of the AST and KPS RAA is not an initiator of a design basis accident. While the proposed amendment would revise certain performance requirements, it does not involve any physical modifications to the plant. Therefore, the proposed amendment does not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, changes in operability requirements during the specified conditions will not significantly increase the probability of occurrence of an accident previously analyzed. Since design basis accident initiators are not being altered by the proposed amendment, the probability of an accident previously evaluated is not affected.

The proposed amendment does not impact the condition or performance of any plant structure, system or component. The proposed amendment does not affect the initiators of any previously analyzed event or the results of mitigation of accident or transient events.

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 amendment does not involve a physical alteration of the plant. No new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed amendment. Similarly, the proposed amendment would not physically change any structures, systems or components involved in the mitigation of any accidents. Thus, no new initiators or precursors of a new or different kind of accident are created. Furthermore, the proposed amendment does not create the possibility of a new accident as a result of new failure modes associated with any equipment or personnel failures.

No changes are being made in the methods used to respond to plant transients that are not addressed in the revised RAA. No changes are being made to parameters within which the plant is normally operated, or in the setpoints which initiate protective or mitigative actions, and no new failure modes are being introduced.

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

   Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The analyses have been performed using conservative methodologies, as specified in RG 1.183. The dose consequences due to design basis accidents are within the applicable acceptance criteria and the guidance of RG 1.183.

   The proposed amendment is associated with the implementation of a new licensing basis for the KPS DBAs. KPS previously obtained NRC approval to use AST methodology as described in RG 1.183. Although a complete revision of the KPS RAA of record has been performed, the proposed amendment continues to ensure that the doses at the exclusion area boundary (EAB) and low population zone boundary (LPZ), as well as the Control Room, continue to be within applicable acceptance criteria.

   The proposed amendment does not impact station operation or any plant structure, system or component that is relied upon for accident mitigation under the revised RAA.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, DEK 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

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. In the AEC's 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."

Because KPS was constructed pre-GDC, the numbering of the KPS GDCs differs from the current GDCs found in 10 CFR 50, Appendix A. As such, the appropriate 10 CFR 50, Appendix A, General Design Criteria are listed below with the associated criteria KPS was licensed to from the Final Safety Analysis (Amendment 7), which has been updated and now titled the Updated Safety Analysis Report (USAR).

Regarding changes proposed in accordance with TSTF-448, DEK makes the following statement: A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on January 17, 2007 (72FR2022), the NRC Notice for Comment published on October 17, 2006 (71 FR 61075), and TSTF-448, Revision 3.

KPS GDC 1 & 5 - Quality Standards and Records (10 CFR 50, Appendix A, GDC 1)

GDC 1 - Quality Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability or codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

The systems and components of the facility have been classified according to their importance in the prevention and mitigation of accidents, which could cause undue risk to the health and safety of the public. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of an accident or result in an uncontrolled release of substantial amounts of radioactivity are designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to reactor operation or safety are designated Class III. These classifications are described in Updated Safety Analysis Report (USAR) Appendix B.

KPS USAR Appendix B lists the Containment Structure (including all penetrations, air locks, isolation valves, vacuum relief devices), the Auxiliary Building (areas housing Auxiliary Building Special Ventilation System, radwaste storage, and Engineered Safety Features) and the Control Room as a Class I structures. Appendix B identifies the following systems; Shield Building Ventilation System, Auxiliary Building Special Ventilation System, and Control Room Air Conditioning and Ventilation System as a Class I systems. Quality standards of material selection, design fabrication and inspection conform to the applicable provisions of recognized codes and good nuclear design practice.

KPS GDC 5 - Records Requirements

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

DEK maintains records of the design, fabrication, construction, and testing of Class I plant components throughout the life of the plant. Additionally, written records are kept of all plant operations, major maintenance, incidents and accidents, and radiation exposure of all personnel and are retained in accordance with the TS and the Operational Quality Assurance Program Description.

KPS GDC 3 - Fire Protection (10 CFR 50, Appendix A, GDC 3)

The reactor facility shall be designed to minimize the probability of events such as fires and explosions, and to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

The KPS Fire Protection Program was developed in accordance with the guidance of Appendix A to Branch Technical Position (BTP) APCSB 9.5-1 as described in NRC's Fire Protection Safety Evaluation Report dated December 12, 1978 (reference 29) and supplement dated February 13, 1981(reference 30). KPS complies with the applicable sections of 10 CFR 50 Appendix R, as described in the Safety Evaluation Report dated December 22, 1981 (reference 31).

Structures, systems and components important to safety are designed and located to minimize the fire hazard. Fire Protection systems are designed to minimize the effects of fires on systems, structures and components important to safety. Adequate means are provided to mitigate the fire hazard encountered in the plant.

Non-combustible and fire resistant materials are used wherever practical throughout the CRE and three-hour rated fire barriers are used to isolate the control room from other areas. Penetrations in fire barriers, such as doorways, cable tray or conduit penetrations, and ventilation penetrations are protected as required. The control room is equipped with portable fire extinguishers, and hose stations are available from adjacent areas.

A dedicated shutdown panel is provided outside the CRE to assure safe shutdown can be achieved should a postulated exposure fire require the evacuation of the control room.

KPS GDC 40 - Missile Protection (10 CFR 50, Appendix A, GDC 4)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

All systems and components designated Class I are so designed so that there is no loss of function in the event of the Design Basis Earthquake acting in the horizontal and vertical directions simultaneously. In addition, all Class I structures are designed to withstand all environmental factors including tornadoes. The working stresses for both Class I and Class II items are kept within code allowable values for the Operational Basis Earthquake. Similarly, measures were taken in the plant design to protect against high winds, flooding, and other natural phenomena. All engineered safety features are protected against dynamic effects and missiles resulting from equipment failures.

KPS is considered to be in full compliance with the KPS GDC-40 as it relates to the Class I structures and Class I equipment.

KPS GDC 4 - Sharing of Systems (10 CFR 50, Appendix A, GDC 5)

Reactor facilities shall not share systems or components, unless it is shown the sharing does not impair safety.

Analyses confirm that the sharing of components among systems does not result in interference with the basic function and operability of these systems and, hence, there is no undue risk to the health and safety of the public. Those systems or components, which are shared functionally within the plant, are designed in such a manner that the sharing does not impair plant safety. Also, KPS is a single-unit site and therefore, there are no shared systems between units.

KPS GDC 10 – Containment (10 CFR 50, Appendix A, GDC 16)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

The total containment consists of two systems:

1. The Primary Containment System consists of a steel structure and its associated-engineered safety features (ESF) Systems. The Primary Containment System, also referred to as the Reactor Containment Vessel, is a low-leakage steel shell, including all its penetrations, designed to confine the radioactive materials that could be released by accidental loss of integrity of the Reactor Coolant System pressure boundary. Systems directly associated with the Primary Containment System include the Internal Containment Spray, Containment Air Cooling and Containment Isolation Systems.

   The principal post-accident function of the Internal Containment Spray and the Containment Air Cooling Systems is to reduce the pressure (and temperature) in the Reactor Containment Vessel.

   The principal function of the Containment Isolation System is to confine the fission products within the Primary Containment System boundary.

2. Secondary Containment System consists of two structures and their associated ESF Systems: the Shield Building and associated ESF System, the Shield Building Ventilation System; and the Auxiliary Building Special Ventilation Zone and associated ESF System, the Auxiliary Building Special Ventilation System. The entire envelope that comprises the Shield Building boundary has been constructed to minimize air leakage across the boundary. The Shield Building concrete structure surrounding the Reactor Containment Vessel is designed to provide:
   - Personnel shielding from the RCS and support systems located inside the Reactor Containment Vessel during both normal operation and Design Basis Accident conditions.
   - Protection of the Reactor Containment Vessel from low temperatures, and other adverse atmospheric conditions, and external missiles.
   - A means for collection and filtration of fission-product leakage from the Reactor Containment Vessel following the DBA. The Shield Building Ventilation System is the ESF system that filters the fission product leakage collected in the Shield Building.

KPS GDC 11 - Control Room (10 CFR 50, Appendix A, GDC 19)

This facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

The control room contains all controls and instrumentation necessary for operation of the reactor, turbine generator, auxiliary and emergency systems under normal or accident conditions. The control room is designed and equipped to minimize the possibility of events, which might preclude occupancy. In addition, provisions were made for bringing the plant to and maintaining a hot shutdown condition from a dedicated shutdown panel located in the turbine building safeguards alley area.

The employment of non-combustible and fire retardant materials in the construction of the control room and the equipment and furnishings, contained therein, minimizes the probability of a control room fire. The location of firefighting equipment in the control room, and the continuous presence of an operator trained to work in smoke with air pack breathing apparatus, and trained in firefighting techniques further reduces the probability that the control room will become uninhabitable.

The revised RAA provided in Attachment 4, demonstrates that the control room is maintained as a safe environment, that access is permitted in a post-accident condition and that radiation exposures to occupants within the control room are maintained below the limits specified in 10 CFR 50.67, which is applicable for plants that have adopted the Alternative Source Term methodology.

KPS GDC 17- Monitoring Radioactivity Releases (10 CFR 50, Appendix A, GDC 64)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

The facility contains means for monitoring the containment atmosphere, effluent discharge paths, and the facility environs for radioactivity, which could be released under any conditions. The details of the effluent discharge path and containment monitoring methods are described in Chapter 11 of the KPS USAR. Some of the details concerning monitoring of radiation (process and area) monitoring have been provided in Section 3.1.4 of this LAR.

Conclusion

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

6.0 ENVIRONMENTAL CONSIDERATION

DEK has reviewed the environmental evaluation included in the model safety evaluation dated January 17, 2007, as part of the CLIIP associated with TSTF-448. DEK has concluded that the staffs findings presented in that evaluation are applicable to KPS and the evaluation is hereby incorporated by reference for this application.

For the changes not included in the CLIIP for TSTF-448, a review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve; (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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. NRC Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000.
2. NRC Generic Letter 2003-01, “Control Room Habitability,” dated June 12, 2003.
3. Letter from Craig W. Lambert (NMC) to Document Control Desk (NRC), “Generic Letter 2003-01: Control Room Habitability – Supplemental Response,” dated April 1, 2005. [ADAMS Accession No. ML050970303]
4. Letter from G. T. Bischoff (DEK) to Document Control Desk (NRC), “License Amendment Request 210 Subject: Technical Specification Modifications Regarding Control Room Envelope Habitability,” dated September 14, 2007. [ADAMS Accession No. ML072620144]
5. TSTF-448, Revision 3, "Control Room Habitability,” dated August 8, 2006. (ADAMS Accession No. ML062210095). Including letter from TSTF to Document Control Desk (NRC), “Corrected Pages for TSTF-448, Revision 3, Control Room Habitability," December 29, 2006. [ADAMS Accession No. ML063630467]
6. Not Used.
7. TSTF-312, Revision 1, “Administratively Control Containment Penetrations,” dated July 16, 1999. [ADAMS Accession No. ML040620147]
8. TSTF-490, Revision 1, “Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec,” dated March 14, 2011. [ADAMS Accession No. ML110730473]
9. Not Used.
10. TSTF-51, Revision 2, "Revise Containment Requirements during Handling of Irradiated Fuel and Core Operations," dated July 31, 2003. [ADAMS Accession No. ML040400343]
11. NRC Branch Technical Position ETSB 11-5, “Postulated Radioactive Releases due to a Waste Gas System Leak or Failure,” Revision 0, 1981.
12. Not Used.
13. Letter from D. V. Pickett (NRC) to J. A. Spina (Constellation Energy), “Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 – Amendment RE: Control Room Habitability (TAC Nos. MD 5928 and MD5929),” dated July 29, 2008. [ADAMS Accession No. ML082030173]
14. Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988.
15. 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Criterion 19, “Control Room (GDC 19)”.
16. Memorandum from C. Craig Harbuck (NRC) to Timothy J. Kobetz (NRC), “Model Application for TSTF-448, Control Room Habitability, Revision 3,” dated February 2, 2007. [ADAMS Accession No. ML070330657]

17. Letter From John Lamb (NRC) to Tom Coutu (NMC), "Kewaunee Nuclear Power Plant - Issuance of Amendment Regarding Implementation of Alternate Source Term (TAC No. MB4596)," dated March 17, 2003. [ADAMS Accession No. ML030210062]

18. Letter from John Lamb (NRC) to Tom Coutu (NMC), "Kewaunee Nuclear Power Plant - Issuance of Amendment Regarding Stretch Power Uprate (TAC No. MB9031)," dated February 27, 2004. [ADAMS Accession No. ML040430633]

19. Kewaunee Power Station, Updated Final Safety Analysis Report, Revision 22, dated March 2010.

20. NUCON Report entitled "Control Room Habitability Tracer Gas Leak Testing at the Kewaunee Nuclear Plant," dated January 27, 2005.

21. Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (NRC), Response to NUREG 0737, Item III.D.3.4, "Control Room Habitability Requirements," dated April 24, 1981.

22. Letter from M. E. Warner (NMC) to Document Control Desk (NRC), "Revision to the Design Basis Radiological Analysis Accident Source Term," dated March 19, 2002. [ADAMS Accession No. ML020870565]

23. Letter from L. N. Hartz (DEK) to Document Control Desk (NRC), "License Amendment Request 211, 'Radiological Accident Analysis and Associated Technical Specifications Change'," dated January 30, 2006. [ADAMS Accession No. ML060540217]

24. Letter from R. F. Kuntz (NRC) to D. A. Christian (DEK), "Kewaunee Power Station - Issuance of Amendment RE: Radiological Accident Analysis and Associated Technical Specifications Change (TAC No. MC9715)," dated March 8, 2007. [ADAMS Accession No. ML070430017]

25. Letter from Steven A. Varga (NRC) to C. W. Giesler (WPSC), "NUREG 0737 Item Number III D.3.4, "Control Room Habitability," dated July 7, 1983.

26. Not Used

27. Not Used.

28. NRC Regulatory Information Summary 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," dated October 18, 2001.

29. Letter from A. Schwencer (NRC) to E. W. James (WPS), dated December 12, 1978.

30. Letter from S. A. Varga (NRC) to Eugene Mathews (WPS), dated February 13, 1981.

31. Letter from S. A. Varga (NRC) to Eugene Mathews (WPS), dated December 22, 1981.